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PUBLIC HEALTH RISKS ASSOCIATED
WITH THE CANDU NUCLEAR FUEL
CYCLE - RADIOLOGICAL RISKS

by

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ABSTRACT

This report analyzes in a preliminary way the risks to the public posed by the CANDU nuclear fuel cycle. Part 1 considers radiological risks, while part 2 evaluates non-radiological risks. The report concludes that, for radiological risks, maximum individual risks to members of the public are less than 10^{-5} per year for postulated accidents, are less than 1% of regulatory limits for normal operation and that collective doses are small, less than 3 person-sieverts. It is also concluded that radiological risks are much smaller than the non-radiological risks posed by activities of the nuclear fuel cycle.

RÉSUMÉ

Le présent rapport présente une analyse préliminaire des risques que le cycle du combustible nucléaire CANDU comporte pour le public. La Partie 1 traite des risques radiologiques, alors que la Partie 2 examine les risques non radiologiques. Le rapport propose en conclusion que dans le cas des risques radiologiques, les risques maximaux pour une personne du public sont inférieurs à 10^{-5} par année en ce qui a trait aux accidents hypothétiques, inférieurs à 1 pour 100 des limites réglementaires associées à l'exploitation courante, et que les doses collectives sont faibles, c'est-à-dire inférieures à trois personnes-sievert. On y conclue, en outre, que les risques radiologiques sont beaucoup plus faibles que les risques non radiologiques qui résultent des activités liées au cycle du combustible nucléaire.

DISCLAIMER

The opinions expressed in this report are those of the authors and do not necessarily reflect those of the Atomic Energy Control Board

FOREWORD

This report is the result of a research project undertaken by contract with the Atomic Energy Control Board on Public Health Risks associated with the CANDU nuclear fuel cycle.

The initial proposal, presented in December 1980, had a wide scope : the whole fuel cycle was to be considered, from uranium ore mining to waste disposal; both radiological and non-radiological hazards were to be assessed; risks would have been separately accounted for according to the nature of the hazard, the severity of the harm, the probability of accidents and the reliability of data. The initial duration of the contract was one year.

As soon as the contract was awarded (February 1981), a working team was set up composed of Dr. W. Paskievici, project director and principal investigator, Dr. L. Zikovsky, co-investigator, Dr. A. Danilov, project engineer and coordinator, and by three M.A.Sc. students, Messrs. D. Brisette, G. Mihelich and A. Ouellet.

The work had been organized as follows : Dr. Zikovsky was in charge of assessing the non-radiological risks; Dr. Danilov of evaluating and selecting the material relevant to the project, and of drafting a background paper on process description of the CANDU fuel cycle; Messrs. Brisette, Ouellet and Mihelich of drafting background papers on methodologies used and results obtained in assessing radiological hazards, in uranium mining and milling, in power plant operation under normal conditions, and in managing spent fuel, respectively : Dr. Paskievici, of developing the methodology of risk evaluation, of reviewing the literature, of establishing contacts with Canadian scientists working in the field of public risks from nuclear industry and of assessing radiological risks from nuclear power plant accidents.

It soon became apparent that the initial project was over-ambitious. A relatively light slippage in the work schedule during the first quarter (February-April) became severe during the second quarter (May-July) when the availability of the project director sharply decreased on account of its nomination as Director of the Nuclear Engineering Institute. However, work was advancing and three background papers were issued for comments during the third quarter (August-October) : "Hazards Associated with Chemical Releases from CANDU Nuclear Fuel Cycle", by Dr. L. Zikovsky, "Hazards Associated with Routine Radioactive Releases from Nuclear Power Plants" by A. Ouellet, "Hazards Associated with Nuclear Waste Disposal" by G. Mihelich and "Process Description of the CANDU Fuel Cycle - Part A : The Front End" by Dr. A. Danilov. Unfortunately, Messrs. Ouellet and Mihelich abandoned the team after producing the first draft and were no longer available. Similarly, the contract with Dr. A. Danilov terminated during this period; however he was able to prepare, later on, a draft on "Process Description of the CANDU Fuel Cycle - Part B : The Power Production". The last background paper, "Hazards Associated with Uranium Mining and Milling" by D. Brissette was completed in January 1982, at which time he also left École Polytechnique. By then, the Project Director had the additional charge of directing the Nuclear Analysis Group and was in total impossibility of pursuing the project.

To prevent total collapse of the project, an agreement was reached between AECB and the Project Director, in December 1981, to reduce the scope of the project and extend the deadline. No additional funds were available for additional manpower. The final report was to be a document which "describes the work done to date in the area of public risk evaluation, identifies areas which require further work and evaluates the quality of the work published to date". The new deadline was set for May 1, 1982, then extended to September 1, 1982. Unfortunately, there has been no progress made during this period. The

situation worsened further when the Project Director was nominated Dean of Research, a full-time position, far from the nuclear area (October 1982). When a new deadline (March 1, 1983) was not respected, it became obvious that it was pointless to try to pursue the project; it was then mutually agreed to terminate it by June 12, 1983.

Thus, the present report summarizes the work done by the principal investigator in the field of assessing radiological risks for the public from the nuclear fuel cycle. It necessarily reflects the change in working conditions and the narrowing of scope. There has been no updating of material, lastly reviewed during the summer of 1981 and, most unfortunately, no further contacts with Canadian experts actively involved in work related to assessing radiological risks.

It is hoped, however, that the present work may be of some usefulness for other investigators, less ambitious but more effective.

Work done by the co-investigator, Dr. Zikovsky, on non-radiological hazards, is presented in a separate cover.

TABLE OF CONTENTS

	Page
FOREWORD	i
CHAPTER 1. <u>INTRODUCTION</u>	1.1
CHAPTER 2. <u>RISK ANALYSIS APPLIED TO RADIOLOGICAL HAZARD</u>	2.1
2.1 Basic Concepts	2.1
a) Individual doses	2.1
b) Collective doses	2.2
c) Health Effects	2.4
2.2 Purpose of Risk Analysis	2.4
2.3 Methodology of Risk Analysis	2.5
2.4 Methodological Problems	2.9
a) Geographical limits	2.9
b) Time limits	2.10
c) Addition of risks	2.11
2.5 Practical Problems	2.12
a) Source models	2.12
b) Dispersion models	2.13
c) Biological models	2.13
d) Demographic models	2.14
2.6 Normalization of Results	2.14
CHAPTER 3. <u>DESCRIPTION OF THE CANDU NUCLEAR FUEL CYCLE</u>	3.1
3.1 Uranium Mining	3.1
a) Process emission points of radionuclides in open-pit mining	3.1
b) Process emission points of radionuclides in underground mining	3.2
3.2 Uranium Mining	3.2
3.3 Uranium Refining	3.3
3.4 Conversion to UF ₆	3.3
3.5 Fuel Fabrication	3.3
3.6 Power Production	3.4

3.7	Reactor Waste Disposal	3.5
a)	Low-level wastes	3.5
b)	Intermediate-level wastes	3.5
c)	High-level wastes	3.5
CHAPTER 4. <u>MINING AND MILLING</u>		4.1
4.1	Introduction	4.1
4.2	Radioactive Materials in the Uranium Ore	4.1
4.3	Emission Sources from Uranium Mining	4.10
a)	Emission sources from open-pit mines	4.10
b)	Emission sources from underground mines	4.11
4.4	Radioactive Materials in the Milling Process	4.13
4.5	Airborne Radioactive Dusts from Mill Process	4.16
a)	Ore dusts	4.16
b)	Yellowcake dusts	4.17
c)	Tailings re-suspension	4.18
4.6	Liquid Effluents from Mill Processes	4.22
a)	Mill effluents	4.23
b)	Tailing effluents	4.25
c)	Seepage	4.28
4.7	Tailings	4.28
4.8	Gaseous Effluents	4.30
4.8.1	Introduction	4.30
4.8.2	Radon emission sources	4.34
a)	Mine ventilation	4.34
b)	Ore pile emanation	4.35
c)	Mill ventilation exhaust	4.35
d)	Tailings emanation	4.35
e)	Emanation from dispersed radium-bearing materials .	4.37
4.9	Health Hazards Due to Radon	4.37
4.10	Collective Doses	4.38
CHAPTER 5. <u>REFINING AND FUEL FABRICATION</u>		5.1
CHAPTER 6. <u>NUCLEAR POWER PLANT OPERATION</u>		6.1
6.1	Introduction	6.1
6.2	Risk from Normal Operation	6.1
6.2.1	Source terms	6.1



6.2.2	Dispersion models	6.2
6.2.3	Collective doses	6.15
6.2.4	Comparisons between models	6.15
6.3	Risks from Accidents	6.17
6.3.1	Introduction	6.17
6.3.2	Safety Systems	6.18
a)	Group one systems	6.19
b)	Group two systems	6.20
c)	Shutdown system No. 1	6.21
d)	Shutdown system No. 2	6.21
e)	Emergency core cooling system (ECGS)	6.21
f)	Containment system	6.25
g)	Emergency water supply	6.27
h)	Emergency power supply	6.27
6.3.3	AECB Safety Objectives	6.27
6.3.4	Safety Performance of CANDU Reactors	6.28
6.3.5	Relative Importance of Human Errors	6.33
6.3.6	Credibility of Results in Risk Studies	6.33

CHAPTER 7. WASTE MANAGEMENT 7.1

7.1	Introduction	7.1
7.2	Reactor Wastes	7.1
7.2.1	Storage	7.1
7.2.2	Processing	7.2
7.2.3	Transportation	7.2
7.3	Irradiated Fuel	7.3
7.3.1	Storage	7.5
7.3.2	Transport	7.5
7.3.3	Disposal	7.6
a)	Concept assessment	7.6
b)	Site selection	7.7
c)	Construction and operation of a demonstration facility	7.7
7.3.3.1	Safety principles	7.7
7.3.3.2	Research and development activities	7.9
a)	Geotechnical research	7.9
b)	Applied chemistry and geochemistry	7.10
c)	Pathway analysis	7.11
d)	Risk assessments	7.11
e)	Comments	7.16
7.3.4	Summary	7.16

CHAPTER 8. CONCLUSIONS 8.1

References	9.1
------------------	-----

LIST OF FIGURES

	Page
1. Risk Assessment Procedure Simplified Diagram	2.8
2. Models and Results Used in Risk Analysis	2.8
3. Complementary Cumulative Distribution Function	2.12
4. Flow Chart of Ore Processing Operating in an Acid Leach Mill	4.2
5. Radioactive Decay Chains for Uranium and Thorium	4.5
6. Summary of Radium and Thorium Balance in Acid Mill Circuits	4.14
7. Summary of Radium and Thorium Balance in Alkaline Mill Circuits	4.15
8. Radioactive Emissions - Pickering GSA	6.3
9. Exposure Pathways to Man	6.11
10. Pathways to Man Considered for Atmospheric Releases ...	6.12
11. Aquatic Pathways to Man	6.13
12. Shutoff Rod	6.22
13. Liquid Injection Shutdown System	6.23
14. Emergency Core Cooling and Recovery System	6.24
15. Reactor Containment System	6.26
16. 1980 Safety System Performance	6.31
17. Probability Distribution for the Whole-body Collective Dose per year	6.35
18. Features Protecting Man from Nuclear Waste	7.8
19. Generalized Flowchart of Possible Exposure Pathways to Man	7.12
20. Post-closure Assessment Methodology	7.13
21. Hydrogeological Modelling	7.14
22. Chemical Modelling	7.15
23. Single Case - Dose VS Time	7.17



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Technologique
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de Montréal

- viii -

24.	Maximum Dose to Most Exposed Individual	7.18
25.	Potential Individual Doses	7.19

LIST OF TABLES

1.	Sources of Contaminants	4.3
2.	Contribution of the Three Radioactive Decay Series to the Total Activity of 1 Metric Ton of Ore Containing 1 kg of Uranium	4.6
3.	Maximum Permissible Concentrations in Air and Water for Members of the Public for Radionuclides of the Uranium-238, -235 and Thorium-232 Decay Chains	4.8
4.	Relative Hazard (Normalized to Radium-226) of the Uranium-238, -235 and Thorium-232 Decay Series	4.9
5.	Comparative Atmospheric Emissions of Radionuclides from Open-pit and Underground Mines for the Recovery of Ore Containing the Equivalent of 1 Ton U_3O_8	4.12
6.	Suspended Particulate Sample Analyses	4.19
7.	Summary of Suspended Particulate Radiological Data	4.21
8.	Dissolved Radium Concentrations in Canadian Mill	4.24
9.	Typical Treated Tailings Pond Effluent Concentrations at Elliott Lake Area (Dissolved Values)	4.27
10.	Radioactivity Delivered to Tailings Areas	4.29
11.	Typical Composition of Elliott Lake Area Tailings	4.31
12.	Radon Daughter Concentrations In and Around Elliott Lake, in W.L.	4.37
13.	Average Atmospheric Releases of Radioactive Effluents from Canadian Nuclear Installations in 1972-1979	6.4
14.	Liquid Waste Discharge from Canadian Nuclear Installations in 1972-1979	6.7
15.	Chronic Atmospheric Releases from a Reference CANDU Reactor	6.10
16.	Chronic Aquatic Releases from a Reference CANDU Reactor	6.10



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Développement
Technologique
École Polytechnique
de Montréal

- ix -

17.	Maximum Boundary Individual Dose Estimate at Reference 4 x 850 MWe Site at Maturity	6.14
18.	Maximum Boundary Individual Dose	6.14
19.	Collective Dose Equivalent Commitments per Unit Emission	6.16
20.	Operating Dose Limits and Reference Dose Limits for Accident Conditions	6.29
21.	Pickering NGS A Result Comparison (Lifetime) System Versus Targets	6.30
22.	Reactor Waste Risks	7.3
23.	Activities of Selected Fission Products and Actinides in Irradiated Fuel	7.4
24.	Irradiated Fuel Risks	7.16

CHAPTER 1. INTRODUCTION

The production of electricity in nuclear power plants involves extraction, processing, transportation and use of radioactive material as well as managing radioactive by-products. During these operations, a fraction of these radioactive substances is released or may accidentally escape into the environment and eventually reach man. Health hazards of humans exposed to radiation are well known at high doses and reasonably well predicted at low doses. Therefore, by careful monitoring radioactive releases which occur during normal operation of facilities along the complete fuel cycle, from ore extraction to waste disposal; by measuring accidental releases, i.e. amounts released per event and frequency of events; by assessing the probability of future accidents as well as their gravity; by modelling the behaviour of radioactive substances in the environment, from the release point to man, (that is by calculating the probable dose exposure, for a given individual, to radiation originated from the radioactive elements present in the original release and from new radioactive elements produced by decay from the initial ones); by using population distribution patterns, to estimate the probable dose exposure to a given population; and by using accepted dose-effect biological relationships, one can, in principle, calculate the public health risks associated with the production of that electricity.

Many studies have been devoted to calculating the radiological hazards arising from part or the entire fuel cycle. The vast majority of these studies applies to the PWR fuel cycle and particularly to USA conditions (v.g. references [1-10]). More recently, results from studies performed in the UK [11], France [12], Federal Republic of Germany [13], USSR [14] as well as in other countries, have been published.

Critical analyses of these studies (see for instance references [15-17]) have shown that differences in the published results originate from differences between methodologies, assumptions, models and local conditions used by the different authors.

The purpose of the present report is therefore two-fold : to establish the framework by which a systematic risk analysis of public radiological hazards could be carried out, and to identify and assess the work done in Canada, in this area, as related to the CANDU nuclear fuel cycle. Due to particular conditions, described in the Foreword, the second purpose has not been entirely reached.

The material presented in the present report is organized in the following way. Chapter 2 is devoted to the theory of risk analysis as applied to radiological public hazards and discusses controversial issues in this area in some detail; this discussion is intended to help researchers in this field to recognize the advantages and limitations of underlying assumptions, the consequences of using particular physical and biological models and the importance of collecting accurate and relevant data. Chapter 3 briefly describes the CANDU nuclear fuel cycle, the different processes which take place within the cycle and the general nature of radioactive releases occurring during these processes.

The next four chapters deal, each one, with a particular aspect of the fuel cycle : Chapter 4 with the mining and milling of uranium, Chapter 5 with further uranium processing, i.e. refining and fuel fabrication, Chapter 6 with nuclear power plant operation and Chapter 7 with spent fuel and nuclear wastes management. In each of these chapters, the focus will be, successively, on releases (normal and accidental), on their dispersion and on their effects on the population. Finally, in Chapter 8, conclusions and additional comments will be presented.

CHAPTER 2. RISK ANALYSIS APPLIED TO RADIOLOGICAL PUBLIC HAZARDS*

2.1 Basic Concepts

Present knowledge on health hazards resulting from exposure to nuclear radiation is analyzed in ICRP [18], UNSCEAR [19] and BEIR [20] reports. An excellent summary of this knowledge can be found in [21]. In his introductory section, the basic concepts used in computing public radiological health hazards will be defined, and some assumptions made for this computation will be reviewed.

a) Individual doses

The effect of radiation on humans stems from the fact that human tissues absorb part of the ionizing radiation emitted by a radioactive source. Hence, the absorbed dose, D , is a measure of the energy deposited in matter by ionizing radiation. The unit absorbed dose is the gray : one gray corresponds to an absorbed energy of a Joule per kilogram of tissue ($1 \text{ Gy} = 100 \text{ rad}$).

The biological effects on a tissue do not depend solely on the absorbed dose but also on the type of radiation. The ICRP has therefore defined the dose equivalent, H , by the relation

$$H = Q D$$

* Material for this chapter has been taken from [16]

where Q , the quality factor, is to be taken equal to 1 for X-rays, γ rays and electrons, to 10 for neutrons (and protons) and to 20 for α particles (and other multiply charged particles). The unit of dose equivalent is the Sievert (1 Sv = 100 rem).

The biological effects differ from the various tissues of the human body. This has lead ICRP to define the effective dose equivalent, H_E :

$$H_E = \sum_T W_T H_T$$

where W_T is a weighting factor representing the proportion of the biological (stochastic) risk resulting from irradiation of tissue T to the total risk when the whole body is irradiated uniformly, H_T is the dose equivalent in tissue T , and the summation is carried out over the same period for all tissues. The effective dose equivalent concept is used for worker radioprotection purposes. It should be noted therefore that there is no weighting factor for the skin and that the weighting factor for the gonads represent the genetic harm in the first two generations only.*

b) Collective doses

In order to provide a measure of the health detriment in an exposed population, the collective dose equivalent, S , has been defined :

$$S = \int_0^{\infty} HN(H)dH$$

* ICRP equates a serious genetic damage to a fatal cancer.



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- 2.3 -

where $N(H)$ is the number of individuals receiving a dose equivalent in the whole body or any specified organ or tissue in the range H to $H + dH$. Theoretically, the integration limits extends to infinity but in practice a spatial limit is taken beyond which the remaining detriment is judged insignificant.

Similarly, the collective effective dose equivalent, S_E , is defined by

$$S_E = \int_0^{\infty} H_E N(H_E) dH_E.$$

Since effects of radiation extend in time and may affect several generations, we must evaluate a temporal integration of the collective effective dose equivalent rate $\dot{S}_E$ to obtain an evaluation of the health detriment caused by the source of radiation. Thus, the collective effective commitment, S_E^C , is defined by

$$S_E^C = \int_0^{\infty} \dot{S}_E (T) dt.$$

For practical purposes, NEA [22] recommends the integration period to be taken equal to 500 years for computing the regional dose and to 10 000 years for global dose commitment. In addition NEA suggests adding terms accounting for risks from later genetic effects and fatal skin cancers. Thus, total detriment from radiation, S_Y , would be :

$$S_Y = S_E^C + 0.25 S_G^C (\text{gonads}) + 0.01 S_S^C (\text{skin})$$

where the weighting factors have been taken from ICRP publications.

In computing S_E^C it is assumed a constant global population of 10^{10} people with constant habits.

c) Health effects

Application of ICRP risk coefficients would give the following health effects :

- number of fatal cancers = $1.65 \times 10^{-4} S_E + 1.65 \times 10^{-6} S_S + 4.1 \times 10^{-5} S_G$
- number of hereditary effects = $10^{-4} S_G$.

2.2 Purpose of Risk Analysis

For a regulatory body, such as the AECB, the purpose of applying risk analysis to the nuclear fuel cycle is to identify activities, within the cycle, which would require remedial actions. These actions fall into two categories : there are regulations for risk limitation and standards for risk reduction. It is a well-known fact that in Canada, as well as in all other countries which follow the recommendations of the International Commission on Radiological Protection (ICRP), normal operations of a nuclear facility are limited, by regulations, to 0.05 Sieverts for workers and to 0.005 Sieverts for individual members of the public. It is perhaps less known that risk reduction is implemented by applying the ALARA principle which states : "All exposures should be kept as low as reasonably achievable, economic and social factors being taken into account" [18]. As far as risks from accidents, standards exist for nuclear power plant operations [23]* and for transportation [25] and are being developed for waste disposal [26, 27].

* See however the modifications proposed by the Advisory Committee on Nuclear Safety [24].

The level of acceptable risk, as defined by the AECB, corresponds to the probability of a fatality of the order of 10^{-4} to 10^{-5} per person (worker or member of the public) from normal operations and of the order of 10^{-6} per person (member of the public) from accidents in nuclear power plants.

Risk analysis would therefore verify that regulations are being met for each activity within the nuclear fuel cycle and would provide data for cost-benefit analysis of alternative ways of reducing the residual risk from these activities.

2.3 Methodology of Risk Analysis

Risk is generally defined as the number of undesirable events occurring in a given period of time. It is then expressed quantitatively as the product of the magnitude of the consequences that result each time an event occurs, by the frequency of that event :

$$\begin{array}{ccccc} \text{Risk} & = & \text{Frequency} & \times & \text{Magnitude} \\ \text{(Consequences/Year)} & & \text{(Events/Year)} & & \text{(Consequences/Event)} \end{array}$$

The consequences, for radiological hazards, vary from imminent death (generally provoked by loss of the immunity defense systems of the human body) due to extremely high doses of radiation (i.e. greater than 10 Sieverts) to late fatal or non-fatal malignancies (i.e. cancers), for low-dose radiations (i.e. less than 0.01 Sievert). Also genetic effects have to be considered.

Frequencies considered in risk analysis cover quite a large range : from 1 per annum, corresponding to normal operation, to 10^{-6} - 10^{-7} per annum, which represents the lower limit of a credible analysis. Events at the very low end of this range have not been observed;

they are postulated for analysis purposes. Therefore, events, and consequently risks, are usually classified as "actuarial" or "hypothetical" according to whether or not direct statistical data exist to make predictions from past experience.

The population at risk, in this study, are individual members of the public. However, it has also been the practice to consider special groups which, by their proximity to radiation sources or by their condition (e.g. pregnant women and infants), are more susceptible to the radiation. It should also be noted that for effects resulting from releases of long-lived radioactive substances produced along the entire fuel cycle, the population at risk theoretically consist of an indefinite number of future generations.

A fuel cycle is defined by all activities (called "steps") needed to produce energy by that fuel. For the CANDU nuclear fuel cycle, the main steps are :

- a) Mining
- b) Milling
- c) Refining
- d) Fuel fabrication
- e) Transport of fuel
- f) Production of electricity
- g) Storage of spent fuel
- h) Conditioning of spent fuel
- i) Transport of spent fuel
- j) Spent fuel disposal
- k) Waste management.

Steps a) and b) are usually considered together since they are generally performed at the same site, by the same company. Refining means, for the CANDU fuel cycle, obtaining high-grade uranium bioxyde powder from the U_3O_8 yellowcake; it may also mean, for the uranium export industry, production of uranium hexafluoride, UF_6 , the raw material used in enrichment plants. Steps h), i) and j) are at the demonstration stage, while step k) is still at the research and development stage. Finally, waste management may apply to various steps of the cycle.

With each step, several "phases" are associated, indicative of the "history of the facility in which an activity takes place. The most important phases are :

- a) Construction
- b) Operation
- c) Decommissioning.

In principle, risk analysis is performed according to the following procedure, schematically shown in Figures 1 and 2. For each activity, events are classified in two categories : normal events, i.e. events associated with normal operations, and abnormal events or accidents. Both categories of events may result in radioactive releases. A source model will provide information on the amount of each radioisotope which is released. Usually, this information is in curies per year (or per event); sometimes, it is normalized per unit of energy produced, v.g. 1 Gigawatt-year (GW-a). Dispersion models give rise to pathway analysis which determine the fraction of the source which may reach man through air, water and food. Biological models, describing the relationships between intakes and internal dose exposures, and between dose exposures (internal or external) and consequences, are used to calculate individual health hazards. Finally, demographic

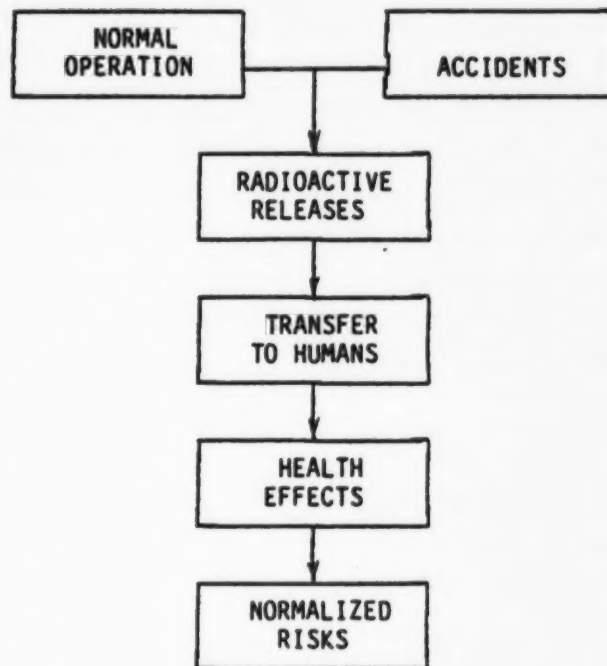


FIGURE 1. Risk Assessment Procedure Simplified Diagram

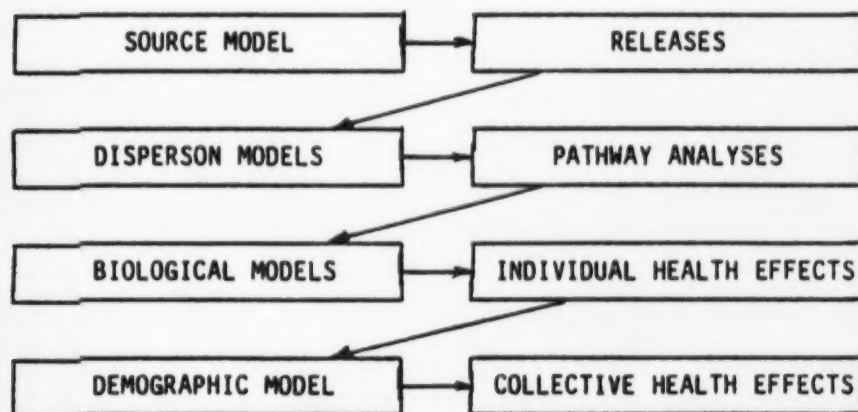


FIGURE 2. Models and Results Used in Risk Analysis

models, representing population distribution around areas affected by radioactive releases, are used to compute effects on the general population. These effects are determined from "collective" dose exposures as well as from "committed" dose exposures, that is, exposures which continue to subsist even after the end of an activity within the cycle. Finally, all health effects are added and often normalized to 1 GW-a.

The addition of health effects is possible, for normal events, because consequences are considered to be proportional with total dose exposures, which means that doses, as well as risks, are additive. However adding risks resulting from accidents of widely different consequences involves a social judgment on the acceptability of these consequences and this judgment is clearly controversial (see section below).

2.4 Methodological Problems

In practice, a number of conceptual and practical problems arise when risk assessment is performed. According to the assumptions made in solving these problems, the results may vary quite widely. It is therefore essential that models, assumptions and methods used in risk analysis should be properly emphasized. This section focuses on conceptual problems.

a) Geographical limits

There is no uniform approach in defining the distances over which effects arising from a source point should be integrated. For radioactive sources, the tendency is to define local (< 80 km), regional (< 2000-3000 km) and global (world-wide) zones, according to the nature of the radioelements and to the physical models used to do pathway analyses. For some authors, only the area in which radioactivity

is measurably higher than the background should be considered; for others, the area should extend to a distance where the computed dose exposure is negligibly small*, or is equal to a small fraction, typically 1%, of the dose exposure at the boundary of the exclusion zone surrounding a nuclear facility; few others insist that the area should encompass the entire surface of the world.

b) Time limits

Theoretically, for radioactive materials dispersed in the geosphere, health effects on populations have to be taken into account until these materials and their radioactive descendants would disappear by decay. The measure of such effects is done via the collective dose commitment concept which is the infinite time integral of the dose rate received by the entire population (present and future generations) from a given source. In practice, several solutions are adopted : a) a time limit of 100 years; this limit combines the simplicity of computation with a span of years for which most people would accept having some obligation for future generations; b) a time limit of 500 years for which most radionuclides would have decayed; this limit is advocated by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR [19]); c) an infinite time limit but ignoring exposures smaller than a "trivial" value; d) an infinite time limit, no limitation on infinitesimal doses; the figures thus obtained would have only a very limited informational value - due to the extreme assumptions used in such models - which should be used only for very specific comparison purposes such as choosing amongst different mill tailing management strategies [28].

* An attempt has been made to define a "trivial" dose, below which effects would be considered; it has been suggested that 1/1000 of the background radiation would be indeed "trivial".

c) Addition of risks

From the conceptual point of view, probably the most difficult problem is the addition of risks. At the heart of this difficulty is the public acceptability of the equivalence between risks which are mathematically equal but which result from completely different situations. Let's first consider two events : one, which occurs annually and results in one death, and the other which would occur once in a century and result in 100 deaths. Both events have the same risk, i.e. one death per year, but one would hesitate to designate them as equivalent, allowing therefore to be added. When more than two events are to be considered - and this is usually done when analyzing different accident sequences -, the events may be summed up, to form narrow-band histograms in a consequence-frequency diagram.

When the number of events to be considered is large, continuous lines, called "complementary cumulative distribution functions" can be used (see Figure 3). The ordinate of a point situated on such a line gives the probability per unit time of having consequences greater than those indicated by the corresponding abscissa. This representation, first used in a reactor safety study [29], appears to be the most appropriate one for representing risks from accidents of wide range of consequences. It should be noted, however, that this sophisticated method of describing risks may be of little use if the maximum dose exposures remain within the stochastic domain*, where effects (number of fatally induced cancers) can be added, according to the linear dose-response biological model. Now, best estimates in reactor safety studies indicate that the probability of accidents resulting in high

* The term "stochastic" is used to describe effects whose probability of occurrence in an exposed population (rather than their severity in an affected individual) is a direct function of dose. The term "nonstochastic" is used to describe effects whose severity is a function of dose.

dose exposures, i.e. outside the stochastic domain, is so low that, for all practical purposes, the radiological risks may be indeed considered additive.

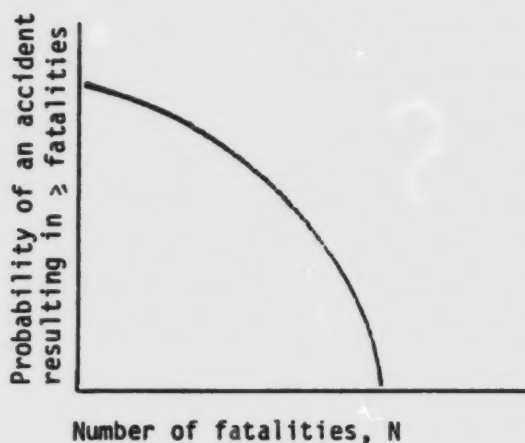


FIGURE 3. Complementary Cumulative Distribution Function

2.4 Practical Problems

The practical problems found in assessing the radiological risks to the public are mainly due to the lack of accuracy in some models and to the reliability of some data used in these models.

a) Source models

Due to a very close monitoring system, releases of radioactive materials from a nuclear installation are very well known for normal operations. However, accidental releases as well as delayed releases from contained wastes (mill tailings, spent fuel vaults, etc.) are more difficult to assess. For licensing purposes, safety reports have to be prepared by the owners of nuclear facilities. These reports, which

have to be approved by the AECB, consider a variety of possible accidents, by assessing the probability of their occurrence as well as the probable consequences. In general, the safety studies are conservative, since not enough information is available on the reliability of complex systems (such as the process and safety systems in a nuclear plant) nor on the long-term behaviour of waste containing means. Failure mode and consequence analysis provides the data needed for developing the source model. Sensitivity studies are sometimes used to assess the uncertainties in either the amount, the frequency of occurrence or the moment of occurrence of unexpected (or unavoidable) releases. These uncertainties strongly vary from one source model to another.

b) Dispersion models

Many models have been developed to describe the dispersion of radioactive isotopes in air, in water, in the ground and in food chains. From simple stationary few-compartmental models, to highly complex, time-dependent multi-compartmental and multi-loop models, the user has a wide range of choices. As the modelling process of physical phenomena increases in accuracy of description, it needs an increasing number of phenomenological parameters whose numerical accuracy is increasingly difficult to ascertain. As the process of model validation is being pursued, the importance of local conditions are increasingly recognized. The practical difficulty is to make the proper choice among the available models and to assess the uncertainties related to that choice.

c) Biological models

Several decades of intensive investigation on the metabolism of different radioisotopes and on the biological effects of ionizing radiation have considerably narrowed the gap in our understanding on these phenomena. The ICRP, UNSCEAR and BEIR reports [18-20] continuously

update our knowledge of the dose-response functions to be used in assessing radiological hazards. As the functions proposed in these reports are very similar (which is quite normal since they have been derived from the same experimental and epidemiological evidence), the only practical difficulty left is the choice between using a single function which would lump somatic and genetic effects, or separately computing these two effects, by taking into account the fact that not all radiation exposure is genetically significant (v.g. α emitters in lung).

d) Demographic models

Demographic models are obtained from national census. Data on distribution and age of population have to be supplemented with data on occupational, food and drinking habits of this population. Practical difficulties arise in obtaining this information and in properly averaging the relevant data.

2.6 Normalization of Results

The radiological hazards to the public resulting from a nuclear activity are presented in the following order, corresponding to an increased complexity in analysis:

a) the maximum radiological risk to an individual member of the public, (in Sv/year)

b) the same risk, but normalized to the production of a given amount of electrical energy, (in Sv/year-GW-a);

c) the collective dose, (in person-Sv/year);

- d) the corresponding normalized risk, (in person-Sv/year-GW-a);
- e) the collective dose commitments, (in person-Sv);
- f) the normalized collective dose commitments, (in person-Sv/GW-a).

For comparative purposes, the results can also be presented as relative figures : either as multiples (or fractions) of the risk due to the background radiation, or as fractions of the risk considered as acceptable by the AECB.

CHAPTER 3. DESCRIPTION OF THE CANDU NUCLEAR FUEL CYCLE

A process description of the CANDU fuel cycle can be found in the Background Paper No. 1 [30], for the front end and for the power production, and in Background Paper No. 4 [31], for the back end; additional details on the front end are given in Background Paper No. 2 [32]. This description is needed in order to identify the process emission points of radionuclides and to provide the necessary information for discussing the accidental releases. In this report, however, only process emission points will be indicated.

3.1 Uranium Mining

The uranium mining is done, in Canada, by both open-pit and underground methods.

a) Process emission points of radionuclides in open-pit mining

The principal source of radioactive effluents of concern in open-pit uranium mining operation is (as indeed in underground mining of uranium) U-238 and its decay products which are in secular equilibrium with its parent in undisturbed geological settings. Present also are the parent radionuclides U-235 and Th-232, but in far less abundant quantities, along with Rb-87 and K-40 in even lesser amounts.

The product of the U-238 decay series consists of some 13 radioactive elements which are shown in the simplified decay sequence of Figure 5. The principal radioactive atmospheric emissions which are in the form of fugitive dust and particulates resulting from mining of the ore zone haulage, stockpiling of ore and waste rock, are : U-238, U-234, Th-230, Ra-226, Pb-210 and Po-210. Also, these radioisotopes are leached by ground water movement through the ore zone and by pre-

cipitation run-off, resulting in radioactive liquid effluents. Atmospheric emission of Radon-222, an inert gas, and its decay products, the Radon daughters (Po-218, Pb-214, Bi-214 and Po-214) are the major airborne radioactive effluents. Although Radon is a gas, its daughters are actually heavy metals which rapidly become attached to aerosols.

b) Process emission points of radionuclides in underground mining

As in open-pit mining, the principal radionuclides of concern are the daughters of the U-238 decay series. These find their way into the mine groundwater as a result of natural leaching action and the air leading to liquid and atmospheric radioactive effluents, respectively. The principal airborne radionuclides, as in open-pit mines, consist of particulates containing U-238, U-234, Th-230, Ra-226, Pb-210 and Po-210; and, Radon-222 gas and its daughters. The same radionuclides are additionally introduced into the mine water due to mining operations.

3.2 Uranium Milling

As in the case of uranium mines, the radionuclides of major concern in the milling operations derive from the U-238 decay series. The mill crushing, grinding product drying and packaging operations account for the great majority of atmospheric radioactive emissions in the form of aerosols and Rn-222 gas. While dust control measures in the form of scrubbers, baghouses and collectors are employed, aerosols containing U-238, U-234, Th-230, Ra-226, Rn-222 (and daughters), Pb-210 and Po-210 are emitted to the atmosphere in small quantities. Since about 10 per cent of the uranium and virtually all of the other members of the uranium series in the original ore are discharged with the tailings (with the only exception of the thorium, up to 5 per cent of which may be retained in the yellowcake, depending on the mill process), the vast preponderance of radionuclides entering the mill are

localized in the tailings impoundment. The dispersion of the above-mentioned radionuclides from tailings storage is by gaseous emission directly to the atmosphere, by wind entrained tailings particles, by seepage to groundwater and controlled liquid discharge to lakes and natural water courses.

3.3 Uranium Refining

Trace amounts of radionuclides comprising the U-238 decay series are found in the off-gas emissions to the atmosphere associated with the operation of the tray dryers and the rotary kiln. Similarly, the high purity ammonium nitrate solution from the centrifuge stage contains traces of unrecovered uranium, and Ra-226.

3.4 Conversion to UF_6

Gaseous emissions to the atmosphere from the UF_6 conversion process contain trace amounts of radionuclides from the U-238 decay chain which originate in the residual amounts found in the uranium trioxide feed. The off-gas emissions associated with the reduction (of UO_3) hydrofluorination (of UO_2), and condensation (of UF_6) are scrubbed with caustic potash solutions to remove hydrogen fluoride gas. The solutions from the scrubbers containing small quantities of entrained uranium are treated for removal of this material which is recycled to solvent extraction circuit in the refining process after chemical treatment to remove its fluoride content.

3.5 Fuel Fabrication

As in the preceding steps of the CANDU fuel cycle, the principal radioactive emissions in the pellet and fuel element fabrication stages come from any residual daughters of the U-238 decay series which

were originally entrained in yellowcake but were not eliminated (recycled, precipitated or otherwise removed) in the subsequent refining and conversion to UO_2 processes. In the fuel producing facility such radioisotopes as : U-234, Th-230, Ra-226, Pb-210 and Po-210, plus Ra-222 gas and its daughters will be found as emissions to the atmosphere and liquid effluent in minute quantities where screening and blending of UO_2 powder is done, during pressing and granulating stages, and when grinding, washing and drying sintered pellets.

3.6 Power Production

Over 99 per cent of all radioactivity generated by a CANDU power plant is formed by fission in the fuel. All but a minute portion of these radioactive fission products remains contained within the fuel element sheating and is eventually stored in spent fuel storage bays at the plant site. Provided the fuel element remains intact, there will be no escape of radionuclides. There occurs, in the course of normal reactor operation, a very small number of fuel element sheath ruptures - about 0.003 per cent. Consequently, during normal operation a small quantity of fission products enters into the coolant and spent fuel bay water, from which it has to be recovered and isolated. In spite of this procedure, gaseous emissions of fission products cannot be entirely avoided. The principal radionuclides involved are : Carbon-14, Iodine-131, Xenon-133, Krypton-85, and less important species.

Another and relatively smaller source of radioactivity is due to activation products, that is nuclides rendered radioactive as a result of neutron capture. These are formed in the heavy water coolant and moderator circuits and comprise Cobalt-60, Cobalt-58, Zinc-65, Iron-59, Manganese-54, and Tritium. In general, even after filtering

and ion-exchange purification, station liquid effluent will contain traces of these radionuclides.

3.7 Reactor Wastes Disposal

a) Low-Level wastes

The low-level solid wastes are a mixture of contaminated paper and plastic trash plus discarded equipment and supplies. Cesium-137 and Cobalt-60 are the most important radionuclides. Because of their low radioactivity, these wastes do not require shielding nor cooling during normal handling and transportation.

b) Intermediate-level wastes

The bulk of intermediate-level wastes is made of spent ion-exchange resins and filters used for the purification of heavy water in reactor and ordinary water from cooling and shielding circuits. The most important radionuclides are, as before, Cesium-137 and Cobalt-60, but many ion-exchange resins also contain appreciable quantities of Carbon-14. These wastes require shielding and adequate confinement but not cooling during normal handling and transportation. Radioactive releases can only occur by accident.

c) High-level wastes

High-level wastes require shielding, adequate confinement and cooling during normal operation. They are made of spent fuel and reprocessed wastes.

The spent fuel contains high activity fission products and transuranic alpha emitters. Reprocessed wastes, considered only as an option in Canada, are those wastes which remain after uranium and plutonium have been extracted from the spent fuel. These wastes, ini-

tially in liquid form, comprise the fission products and the actinides other than plutonium; only about one-half of one per cent of the initial plutonium remains with the reprocessed wastes.

Presently, spent fuel is confined and cooled in a purified water-filled discharge bay on the reactor site. At some future time, plans are to condition it and to ship it to a waste disposal facility where it will be confined in a deep-buried vault. Releases of radioactive materials, except from accidental causes during conditioning and shipping, would only occur by slow failure of the vault, of the spent fuel container and by eventual leaking of fuel.

CHAPTER 4. MINING AND MILLING*

4.1 Introduction

Mining and milling of uranium result in the formation of waste products formed in the various stages of the process (Figure 4) [33]. These affect the environment to varying degrees and they must be suitably disposed of. They consist of solid wastes, mainly from the mining process, liquid wastes which are residues from the milling processes or leakage during the milling, and airborne waste in the form of dust, chemical contaminants and radioactive substances, principally radon (Table 1 [34]).

The final stage in the milling process involves storage of the mill tailings which is potentially the largest source of contaminant loadings into the environment.

Sources of radioactive contaminants released throughout mining and milling processes are reviewed both qualitatively and quantitatively in the next sections.

4.2 Radioactive Materials in the Uranium Ore

Naturally occurring uranium consists of three different isotopes: uranium-238, uranium-235 and uranium-234 which have a relative abundance of 99.3%, 0.7% and about $5 \times 10^{-6}\%$ respectively. Two of these isotopes, U-238 and U-235, are the long-lived parents of long

*The major portion of this chapter is adapted from Chapter 4 of the Background Paper No. 2 [32].

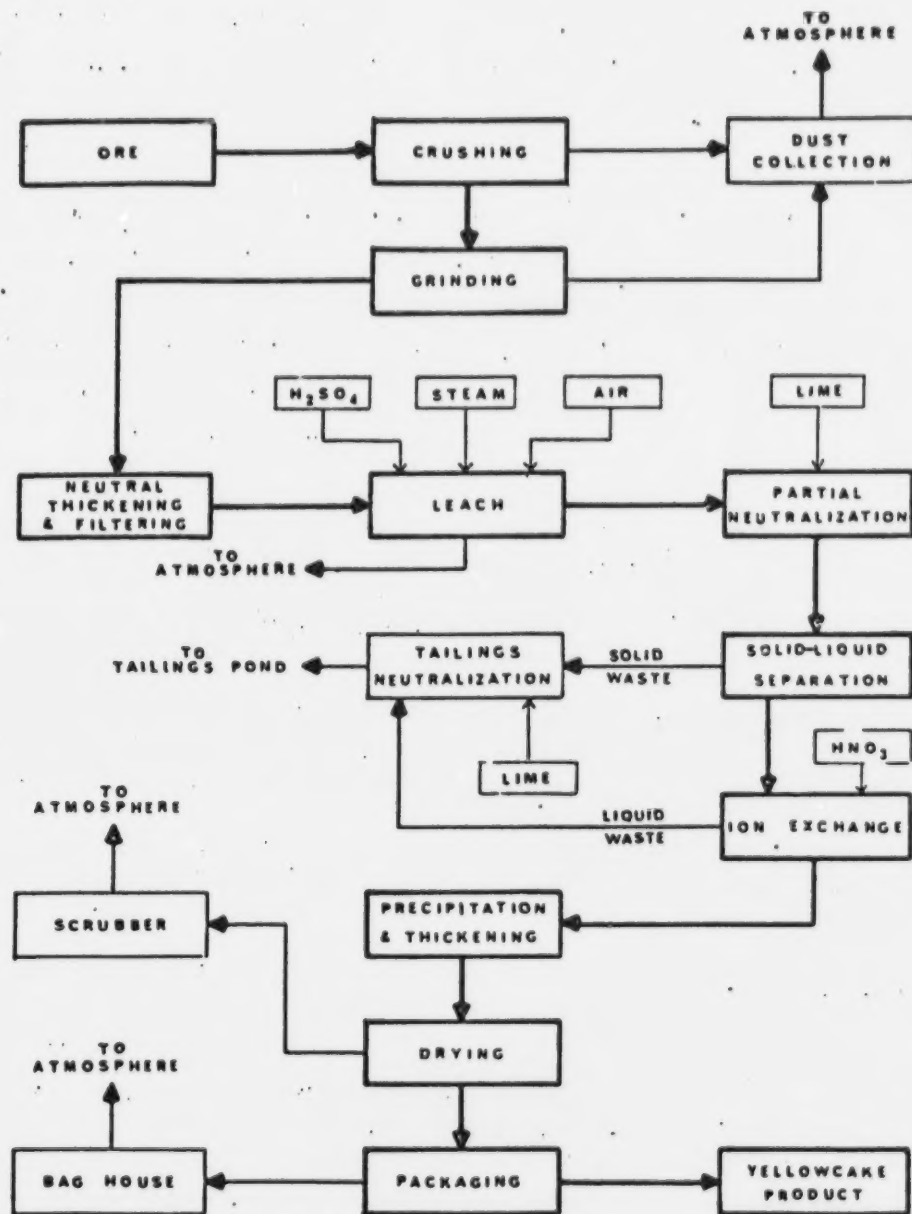


Figure 4 - FLOW CHART OF ORE PROCESSING OPERATING IN AN ACID LEACH MILL

TABLE 1 - SOURCES OF CONTAMINANTS

Source	Contaminants
Waste Rock	Radionuclides, Acid, Heavy Metals.
Minewater	Radium, uranium and other radionuclides, nitrogen compounds, acid, sulphates, iron, heavy metals, dissolved and suspended solids.
Milling	Liberation of radionuclides, addition of N-compounds, sulphuric acid, limestone, lime.
Tailings Area (Active and Inactive)	Dusting and erosion of the surface of the tailings; radionuclides, acid, heavy metals in seepage to surface waters or groundwater. (Radon emanation).
Tailings Area Overflow	Radium and other radionuclides, N-compounds, dissolved solids.
Settling Treatment Pond(s) Overflow	Dissolved and suspended radium, N-compounds, dissolved solids.
Settling Treatment Pond(s) (Inactive)	Radium and barium from redissolution of (Ba,Ra)SO ₄ sludges from treatment of overflow with BaCl ₂ to remove radium.

- 4.4 -

chains of successive radioisotopes called respectively the "uranium series" and the "actinium series". Uranium-234 belongs to the uranium-238 series.

Another radioactive isotope, thorium-238, which is the first member of still another long chain of successive radioisotopes (namely the "thorium series"), is often found associated with uranium bearing minerals.

Uranium, actinium, and thorium series which consist respectively of 14, 11 and 10 radioactive members are shown in Figure 5. The contribution of each decay series to the total activity of metric ton of ore is illustrated in Table 2, assuming an uranium equivalent ore content of one kilogramme and a U/Th ratio of 4:1.

The U-238 isotope is by far the more abundant, in terms of activity, of all naturally occurring radioisotopes in uranium ore and its series contributes to over 90% of the total ore activity. However, radionuclides of primary concern are those which have a high relative hazard. The relative hazard of a given radionuclide can be estimated by dividing the amount of activity (in curies) present in one tonne of ore by the maximum permissible concentration in air or water (MPC_a or MPC_w) of that particular radionuclide. These maximum permissible concentrations, established by the International Commission of Radiological Protection (ICRP), are derived concentration standards which limit doses from continuous exposure to within the prescribed dose limits for the whole body or individual organs or tissues.

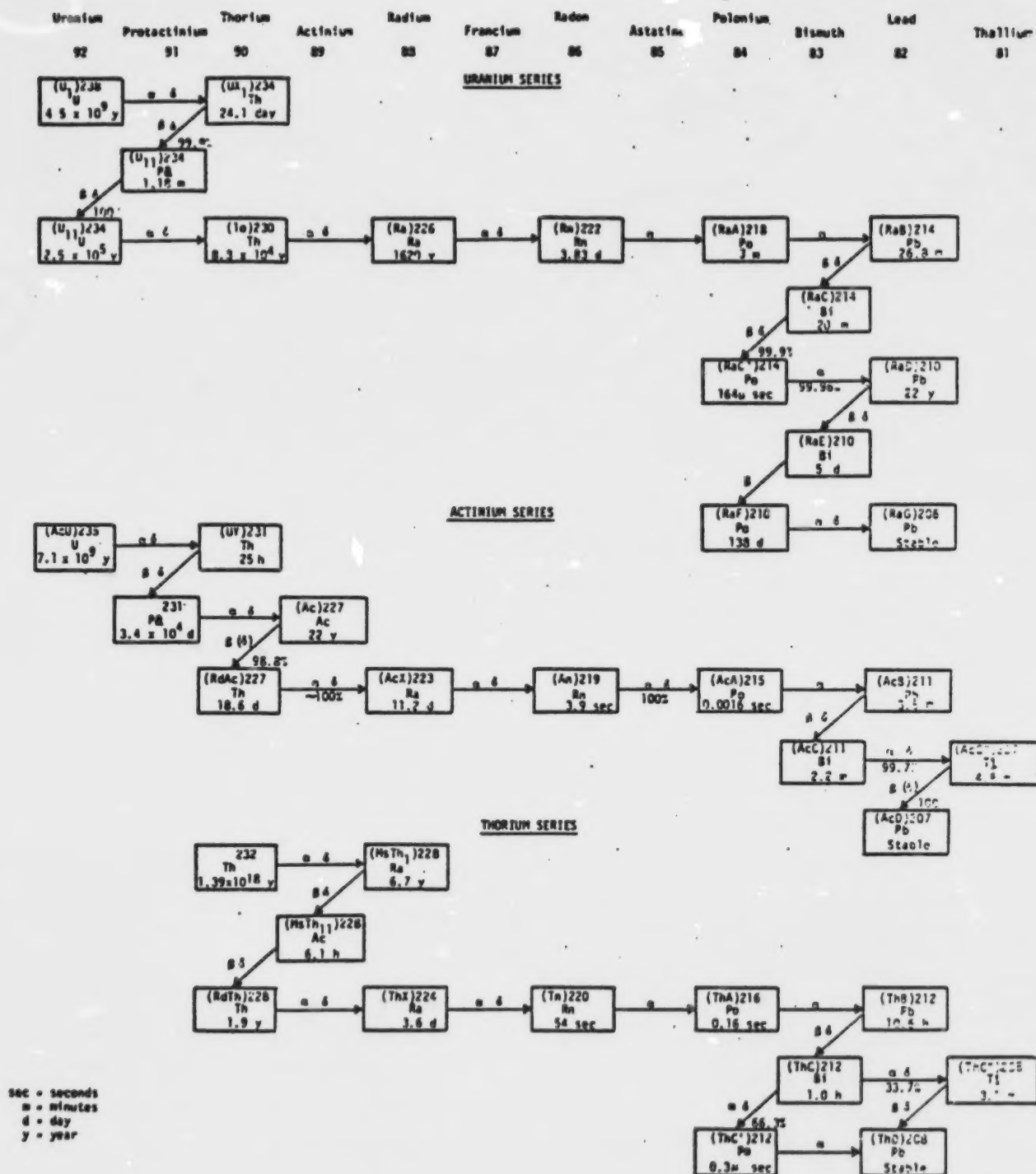


Figure 5

RADIOACTIVE DECAY CHAINS FOR URANIUM AND THORIUM



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Développement
Technologique
École Polytechnique
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- 4.6 -

TABLE 2. CONTRIBUTION OF THE THREE RADIOACTIVE DECAY SERIES TO THE TOTAL ACTIVITY OF 1 METRIC TON OF ORE CONTAINING 1 KG OF URANIUM*

DECAY SERIES	MEMBERS	ACTIVITY OF EACH MEMBER (μCi)	TOTAL ACTIVITY (μCi)	PERCENTAGE OF TOTAL ACTIVITY
Uranium	U-238 + 13 daughters	333	4662	91.5%
Actinium	U-235 + 10 daughters	15	165	3.2%
Thorium	Th-232 + 9 daughters	27	270	5.3%
			<u>5097</u>	<u>100%</u>

- * Assuming:
- secular equilibrium
 - 0.1% U nat in ore (or 0.118% U_3O_8)
 - an uranium to thorium ratio of 4:1.

These assumptions holds for the ore currently mined in the Elliot Lake region and wich accounts for over 70% of the canadian production.

Maximum permissible concentrations in air and water for radionuclides of the U-238, U-235 and Th-232 decay series are and are listed in a decreasing order of importance in Table 3.

It readily appears from relative hazard values listed in Table 4 that it is the U-238 series that contributes to essentially all of the radiological hazard although a few radionuclides of the U-235 series (i.e. Pa-231, Ac-227, U-235) and the Th-232 series (Th-228, Th-232, Ra-228) warrant some considerations.

Selective leaching, diffusion, transport or uptake must equally be taken in consideration when identifying radioisotopes of concern.

For instance radon-222, which is an inert gas with a 3.82 day half-life, is of particular interest even though it shows a low relative hazard when compared to other radioisotopes listed in Table 4.6. Radon is readily released to the air upon fracturing of the mineral crystal. Only a fraction of radon actually present can be released due to trapping in the mineral crystals. The available fraction is generally taken to be 0.29 [35]. Even when all the radon is removed from ore, it will be replaced by decay of radium-226 which has a long half-life (1620 years).

Daughter decay products of radon are solids; when radon is removed as a gas, daughter products (Po-218, Pb-214, Bi-214, Pb-210, Bi-210, Po-210 and Pb-206) will be generated by decay and attach to small particles in the air. Eventually, dry deposition and rain scavenging will bring these residual radionuclides to earth surfaces.

TABLE 3. MAXIMUM PERMISSIBLE CONCENTRATIONS⁽¹⁾ IN AIR AND WATER
FOR MEMBERS OF THE PUBLIC FOR RADIONUCLIDES OF THE
URANIUM-238, URANIUM-235 AND THORIUM-232 DECAY CHAINS

MPC FOR AIR (pCi/m ³)				MPC FOR WATER (pCi/l)			
ISOTOPE	DECAY CHAIN			ISOTOPE	DECAY CHAIN		
	U-238	U-235	Th-232		U-238	U-235	Th-232
Protactinium-231		0.04		Radium-226	10		
Actinium-227		0.08		Radium-228			30
Thorium-230	0.08			Lead-210	100		
Thorium-228			0.2	Uranium-238	600		
Radium-226	1			Polonium-210	700		
Thorium-232			1	Radium-223		700	
Radium-228			1	Protactinium-231		900	
Uranium-238	3			Actinium-227		2000	
Uranium-234	4			Thorium-230	2000		
Lead-210	4			Thorium-232			2000
Uranium-235		4		Radium-224			2000
Thorium-227		6					
Polonium-210	7						
Radium-223		8					
Radium-224			20				
Bismuth-210	200						
Actinium-228			600				
Radon-222	1000						
Thorium-234	1000						

(¹) The most restrictive MPC, soluble or insoluble is listed.

TABLE 4. RELATIVE HAZARD^(*) (NORMALIZED TO RADIUM-226)
OF THE URANIUM-238, URANIUM-235 AND THORIUM-232
DECAY SERIES

	ISOTOPE	RELATIVE HAZARD IN AIR	ISOTOPE	RELATIVE HAZARD IN WATER
	Th-230	13	Ra-226	1
	Ra-226	1	Pb-210	0.1
	U-238	0.33	U-238	0.017
	U-234	0.25	Po-210	0.014
	Pb-210	0.25	Th-230	0.005
	Po-210	0.14	Th-234	0.0005
	Bi-210	0.005		
	Rn-222	0.001		
	Th-234	0.001		
	Pa-231	1.1	Ra-223	0.00064
	Ac-227	0.56	Pa-231	0.0005
	U-235	0.011	Ac-227	0.00023
	Th-227	0.0075	U-235	0.00011
	Ra-223	0.0056		
	Th-228	0.41	Ra-228	0.027
	Th-232	0.08	Ra-224	0.0004
	Ra-228	0.08	Th-232	0.0004
	Ra-224	0.004	Th-228	0.0001
	Ac-228	0.00014		

* The relative hazard of a particular isotope is obtained by dividing the number of curie present in one tonne of ore by the most restrictive MPC for that isotope. A "normalized" relative hazard is obtained by dividing arbitrarily by the relative hazard of Radium-226.

4.3 Emission Sources from Uranium Mining

Sources of radioactivity releases from mining operation have been identified as:

- mine ventilation (radon and airborne radioactive particulates)
- ore stock pile (radon and particulate, leaching)
- mine water.

a) Emission sources from open-pit mines

Radon-222 is believed to be the major airborne radioactive effluent from open-pit mines [36]. Because of the diffuse nature of these mining operations, no direct measurement of Rn-222 total emission can be made; therefore, emissions rates of radon from open-pit mines must be estimated using calculational methods. Atmospheric emissions of Rn-222 from a model U.S. open-pit uranium mine, have been estimated by EPA to be around 63 $\mu\text{Ci/s}$. Radon emission rate from open-pit operations at Cluff Lake's "D" ore body, where a uranium content up to 30% is found, has been calculated to be about 11 microcuries per second [33]. The emission is estimated to be relatively low in spite of the high ore uranium content due to the use of an inert sand shield over the mined area of the pit. Particulate emissions from open-pit mines appear to be much less significant (1/10 000) than Rn-222 [33].

Groundwater seepage into the mine and water from precipitation run off at the open-pit mine will contain leached radionuclides. Composition of mine water varies significantly with mine location and relatively few definitive analysis are available. After treatment, the radon-226 activity is reduced to about 50 pCi/l. [33]

b) Emission sources from underground mines

Radon-222 appears to be the main radioactive effluent in the mine ventilation air. In order to lower the concentration of radon daughters in the air, about $0.1 \text{ m}^3/\text{s}$ of air per ton of ore broken is circulated in underground mines. McLaren [37] has estimated that there is, for Denison and Quirke mines, an average emission rate of 1.0 curie per day of Rn-222.

Although the total quantity of material handled in the mining operations is large, air emissions from these operations are relatively small. Most sources of particulate emission from blasting, drilling, and ore recovery operations are found to be controlled. Water sprays are used to reduce dust levels at work stations. Particulate emission rates for Denison and Rio Algom mines have been calculated and shown to be relatively small.

Atmospheric emissions of radionuclides from open-pit and underground mines generated by the recovery of varying quantity of ore containing an equivalent amount of 1 ton U_3O_8 are compared in Table 5.

Water used in the underground mine for dust suppression, drill flushwater, equipment cleaning, etc., along with groundwater seepage into the mine will contain leached radionuclides. Mine water is pumped to the surface and discharged in tailings ponds or treated for use as processing water in the mill. Typical mine water flows at Denison and Rio Algom are 35 litres/sec [37]. Radioactivity levels can be quite high. Typical concentrations are:

Ra-226 (dissolved)	150-550 pCi/litre
Ra-226 (total)	~750 pCi/litre
U	40-200 mg/litre
Th	10-20 mg/litre

TABLE 5. COMPARATIVE ATMOSPHERIC EMISSIONS OF RADIONUCLIDES
FROM OPEN-PIT AND UNDERGROUND MINES FOR THE RECOVERY
OF ORE CONTAINING THE EQUIVALENT OF 1 TON U_3O_8 ⁽¹⁾

TYPE OF OPERATION	DATA SOURCE	AIR EMISSION SOURCES	
		RADON-222 (Ci)	TOTAL PARTICULATES (Ci)
Open pit ⁽²⁾	(33)	3.0	3.4×10^{-4}
Open pit ⁽²⁾	(21)	3.1	—
Underground ⁽²⁾	(21)	18	—
Underground ⁽³⁾	(8)		
(Denison)		0.8	6.2×10^{-5}
(Rio Algom)		1.0	6.7×10^{-5}

(1) Calculated values.

(2) Estimated values for model mine.

(3) Obtained by assuming 24 hours per day emissions and a 0.1% U_3O_8 ore content.

Because of the unusually high uranium content of the "D" ore pit at Cluff Lake, careful monitoring and treatment must be maintained as mine water activity could be of the order of 20000 pCi/litre of Ra-226 [30].

Mine water is not considered to be a significant source for release of radioactive contaminants into the environment.

4.4 Radioactive Materials in the Milling Process

The uranium extraction process is designed for the maximum possible removal of uranium from the ore. Existing mills operate with an efficiency of greater than 95% uranium extraction. The milling process has no net effect on the total amount of radioactivity which enters the mill each day but causes some redistribution of radioactive materials in addition to recovering the uranium. About 15% of the total radioactivity in the ore entering the mill reportedly leaves in the yellowcake concentrate whereas from 50 to 85% of the original activity is discharged in the tailings, depending upon the proportion of the radon lost. The several radionuclides present at radioactive equilibrium in the uranium decay chain have chemical characteristics different from that of natural uranium parents; most of the radionuclides have been found to remain insoluble during leaching and leave the mill with solids tailings. The amount of radionuclides leached from the ore depends strongly on the type of leaching process in use.

Radium and thorium balance in acid and alkaline mill circuits, as compiled by Itzhovitch[], is summarized in Figures 6 and 7.

Other radionuclides are apparently rejected to tailings; radiation measurements do not indicate their presence in yellow-cake [37].

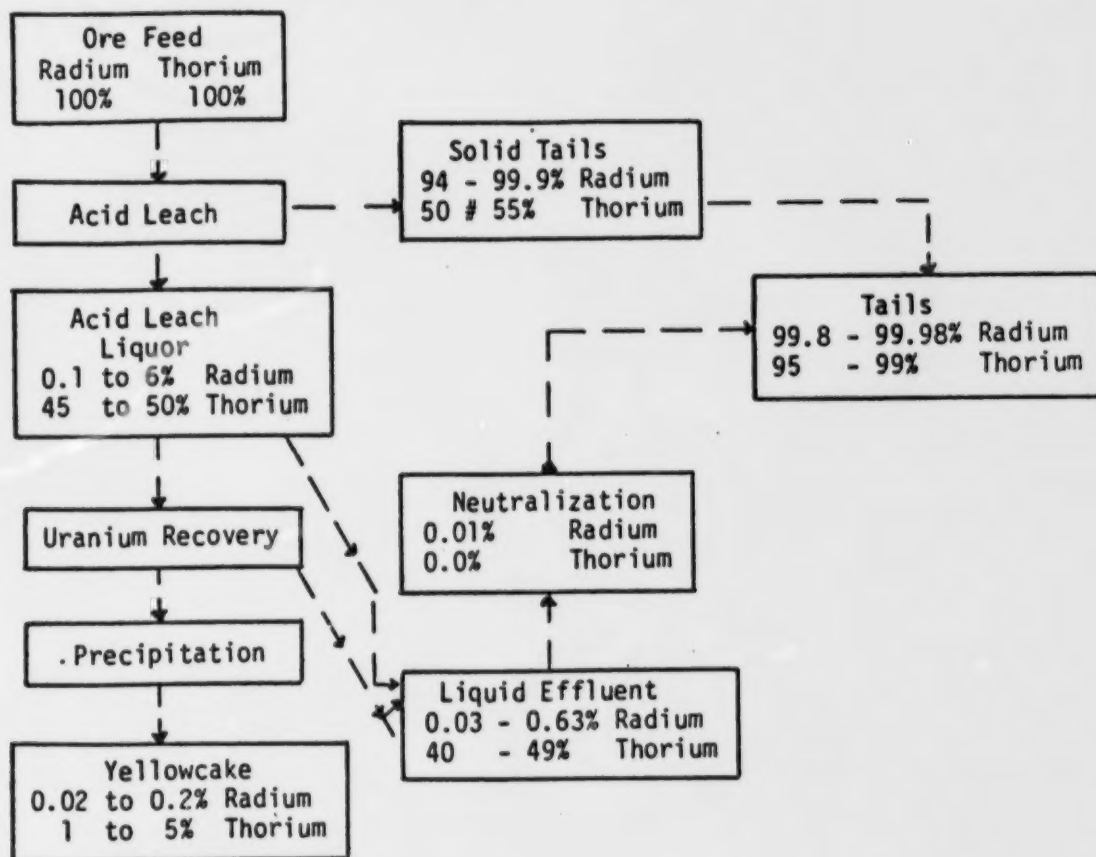


Figure 6 - SUMMARY OF RADIUM AND THORIUM BALANCE IN ACID MILL CIRCUITS

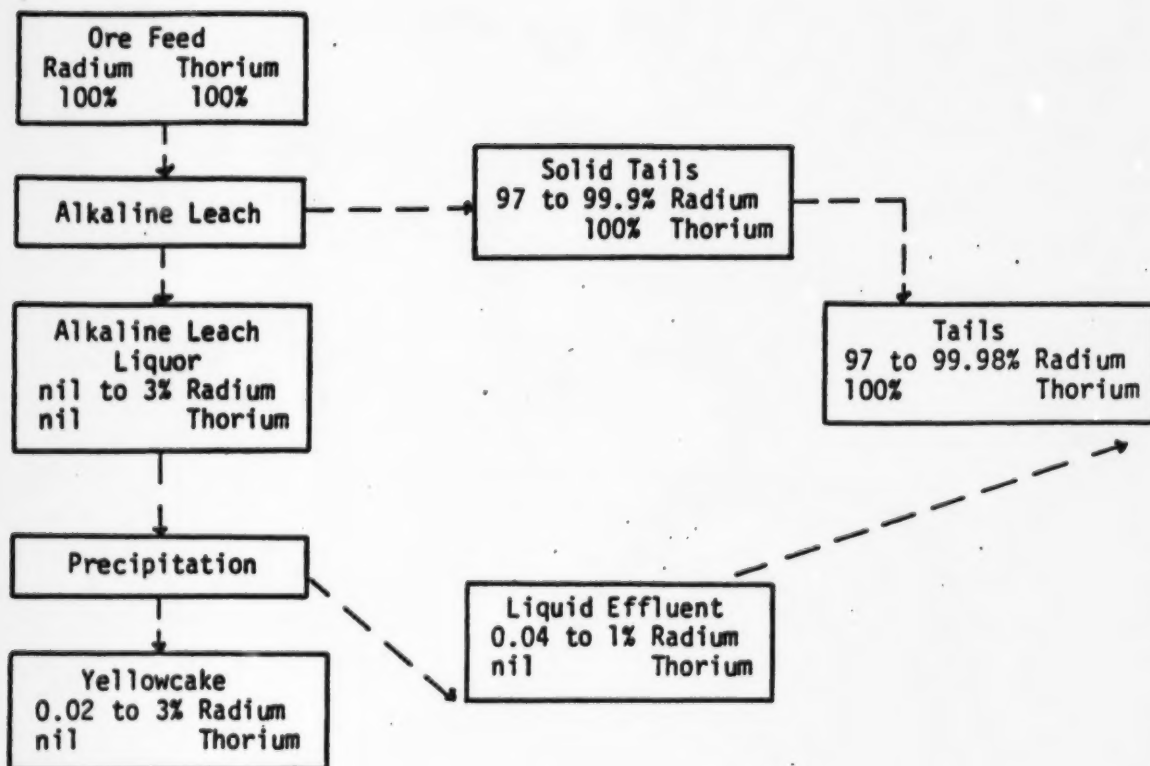


Figure 7 - SUMMARY OF RADIUM AND THORIUM BALANCE IN ALKALINE MILL CIRCUITS

process. No data is available on their concentrations in uranium mill process streams.

4.5 Airborne Radioactive Dusts from Mill Process

Numerous opportunities arise for the formation of airborne radioactive particulates in the milling processes: ore storage handling, crushing, screening, transferring, etc., yellowcake drying and packaging, and tailings disposal. Dust producing activities are essentially the same in all mills and are unrelated to the chemical flow-sheet. Sources of airborne particulate radioactivity are:

- Mine ventilation exhaust
- Mill discharges
- Re-suspension of tailings and radioactive materials on the ground.

a) Ore dusts

Surface crushing, screening and conveyor transfer points in most mills are enclosed and exhausted through fabric filters which remove most of the ore dust entrained in the exhaust air. Fine dusts (<10 microns diameter), in which the uranium is reported to be concentrated, pass through the dust collector. McLaren [37] has estimated the particulate emission rate from these sources, based on gravimetric samples from ventilation exhausts of two mills in the Elliot Lake area, is around 1 g/s, with a radon emission rate of about 60 $\mu\text{Ci/s}$.

Airborne radioactive particulates releases from ore crushing/grinding circuits consists of ore dust containing uranium in radioactive equilibrium with its daughters. Radon emission rate is around 1 $\mu\text{Ci/s}$.

b) Yellowcake dusts

The second important composite source of radioactive emission from mill occurs at the end of the ore processing line: the yellowcake drying and packaging operation.

In most mills, yellowcake drying operation exhausts through a wet scrubber and fan, or a bag filter prior to release. The collected material is returned to the mill circuit. In addition, the yellowcake handling rooms are ventilated in order to protect the health of the workers. The ventilation air from the packaging operation is passed through bag filters prior to release in the atmosphere.

Releases of yellowcake dust from the drying and packaging areas consist of mostly uranium without daughters. This accounts for some 80-90% of the radioactive particulate releases from the mill.

McLaren [37] gives the yellowcake release rate as 0.05 to 0.25 g/sec. An average release rate of 0.1 g/sec. corresponds to 0.034 $\mu\text{Ci/sec}$. of U-234 [33].

c) Tailings re-suspension

The problem of tailings re-suspension by wind erosion exists with dry open tailings areas. It is particularly significant during summer and fall when dry beach areas exist and during gusty wind conditions. The release of particulate material is difficult to quantify, being dependent on how dry the tailings material is, its particle size distribution, and on wind velocity. Maximum winds and gustiness are more important than average wind velocity. Tailings stabilization such as re-vegetation is expected to significantly reduce the problem of wind erosion.

The solid tailings contain more than 60% of the total long-lived radioactivity originally present in the ore, including about 7% of the uranium, and essentially all of the Th-230, Ra-226, Pb-210, and Po-210.

Several suspended particulates samples next to the tailing areas were analysed for their radioisotope content and the results are presented in Table 6. [37].

Results of the air monitoring survey conducted at Rio Algom's Nordic tailings area clearly indicate the effects of annual climatic changes on airborne releases from tailings areas. The arithmetic means, geometric means, and highest values recorded during the winter months are very low. During the spring period, the values increase somewhat, but it is apparent from the data that summer conditions result in higher values. The data indicate that the tailings are not prone to becoming airborne during the winter months, and are less likely to drift in the spring than in the summer. It has been shown [37] that dust entrainment could be limited by frost and/or snow cover approximately 30% of the year.

TABLE 6 - SUSPENDED PARTICULATE SAMPLE ANALYSES

Sample Stations and Dates	Non-Operating Tailings Areas			Operating Tailings Areas		
	Stanrock	Nordic		Quirke	Denison	
	No.1 - June 21	No.1 - July 25		No.1 - July 17, 20, 23, 24, 27	No.1 - Sept. 9 No.2 - Sept. 16 No.3 - Sept. 17	
Particulate Concentrations ($\mu\text{g}/\text{m}^3$)	1050	1970		87 (mean)	206 (mean)	
	HPC _A *	Radionuclide Analyses				
	pCi/m ³	pCi/g	pCi/m ³	pCi/g	pCi/m ³	pCi/g
Lead-210	4	292	0.307	137	0.013	195
Polonium-210	7	222	0.233	33.6	0.008	159
Radium-226	1	N.D.	N.D.	17.9	0.05	7.2
Thorium-228	0.2	1.4	0.001	0.6	0.0001	30.3
Thorium-230	0.08	0.6	0.001	0.7	0.0002	38.3
Thorium-232	1	<0.1	<0.001	0.3	0.001	7.9

* MPC_A - Maximum Permissible Concentration in Air as set by the ICRP

N.D. denotes not detectable.



- 4.20 -

Suspended particulate radiological data from monitoring survey of populated sites in the Elliot Lake area have been reported by McLaren [37] and are presented in Table 7.

A model for dispersion of dust from a uranium mill tailings area has been developed by McLaren [37] in order to estimate the annual distribution of suspended particulate matter and dustfall as it is carried away from seven major tailings sources in the Elliot Lake area and in all compass directions. The model is based on a concept developed by the Oak Ridge National Laboratory and includes particle saltation, suspension and deposition and atmospheric transport using a Gaussian plume model. It was conservatively assumed in this model that tailings which are exposed in the summer remain in the same unfrozen and exposed state throughout the year. The model was calibrated using the data from the suspended particulate and dustfall field sampling programme.

Results of the model were compared with suspended particulate field studies and were found to be in close agreement. Model results indicated that the aerial extent of dust carried from the tailings area is limited. Levels from the larger sources drop off by an order of magnitude within 3 to 4 km.

Suspended particulate and deposition modelling results for Elliot Lake Townsites were compared to actual measured data [37]. It was concluded that the influence of tailings dust in the surrounding townsites is relatively small.

TABLE 7 - SUMMARY OF SUSPENDED PARTICULATE RADIOLOGICAL DATA

Sites	Time Period	Gross Alpha		Gross Beta		Ra-226		Radium		Po-210		Th-232		Th-230		Th-234	
		Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)	Avg. Above D.L. (pCi/m ³)
Ellet Lake (Quirke Township) Benison Township Hardie Township	November to August 30/78	29	0.65	41	0.10	2	0.42	10	0.016	-	-	-	-	-	-	-	-
Ellet Lake (Quirke Township) Benison Township Hardie Township	09/10/77 to 12/31/77	27	0.070	77	0.000	-	-	-	-	-	-	-	-	-	-	-	-
Quirke Township (Benison Township)	01/06/78 to 02/13/78	-	-	-	-	-	-	-	100	0.010	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Ellet Lake (Quirke Township) Benison Township Hardie Township	02/10/77 to 02/13/78	-	-	-	-	14	0.012	26	0.007	-	-	-	-	-	-	-	-

Avg. denotes average
N.D. denotes Not Detectable
D.L. denotes Detection Limit, not presented

4.6 Liquid Effluents from Mill Processes

Liquid wastes originate from the grinding, leaching and extraction circuits of the mill. Large volumes of acidic or basic liquid wastes which contain high concentrations of chemicals in addition to dissolved radioisotopes are generated by the uranium mill. Mill waste liquor is not directly discharged to the environment; it is used in the tailings system as a medium to transport the sand and slimes to the tailings pond.

The mill-process effluents from both the acid and alkaline leach operations are discharged at an alkaline pH and, when combined with the mine water, result in the pumping into the tailings area of about one ton of water for each ton of ore milled.

The discharged liquid is contained in the tailings impoundment where evaporation, seepage and controlled decant outflow occur. In most cases, water from the impoundment is recycled back to the mill.

Process wastes contain the highly soluble reagents added in the mill and the major contaminants in the final effluents from the uranium industry are derived from these added chemicals rather than from the ore. The radioisotope radium-226 is the single exception to this.

Neutralization of the liquid effluents with lime, which is a common practice in Canada, is effective in reducing the concentrations of many objectionable ions as well as eliminating free acid.

Seepage and run-off from idle and abandoned tailings areas present a continuing problem. Currently, this highly polluted water is being treated by lime addition to raise the pH to an acceptable level

and remove the metal cations by their precipitation as hydroxides. The dissolved radium is reduced by precipitation with barium chloride.

Liquid effluents, although containing only a minor percentage of the total activity in the ore, are the major environmental concern for the predominantly wet canadian environmental conditions because tailing effluents are ultimately discharged to natural watercourses.

a) Mill effluents

Radium-226, very early on, was identified as the major isotope of concern for environmental control of liquid discharges from uranium mills. Therefore, the majority of the reported studies, even to this day, are concerned with control of this nuclide. The acid-leach process dissolves a small amount of the total radium present in the ore, but virtually all of this is discharged with the tailings. Neutralization of the tailings slurry to pH 9 and up reportedly precipitates as insoluble oxides or hydroxides about 90% of the dissolved radium and almost all of the dissolved thorium, uranium, and other heavy metal ions.

Reported Ra-226 concentrations in liquid effluents from Canadian mills are summarized in Table 8.

Very little information is available on concentrations of other dissolved radionuclides in canadian mill effluents. However, an analysis of published data on radium and thorium concentrations in U.S. liquid mill effluents shows that concentrations of dissolved thorium shows a clear depression in the higher pH range.

TABLE 8. DISSOLVED RADIUM CONCENTRATIONS IN CANADIAN MILL

MINE/MILL	MILL EFFLUENTS			DATA SOURCE	TAILINGS EFFLUENTS		BEFORE TREATMENT	AFTER TREATMENT	DATA SOURCE
	FLOW RATE (μ /sec.)	pH RANGE	Ra-226 (pCi/ μ)		FLOW RATE (μ /sec.)	pH RANGE			
Rio Algom (Quirke)		10-11	800	(38)	250		400-1000	3-10	(1,34)
Denison	140	9.5	4500	(39)	250		350-1000	2-7	(1,34)
Beaverlodge			70	(35)	70-270		70	3.6	(1,35)
Madawaska	43		2000-7000	(11)	27	7.1	2000	8.5	(11,40)
Gulf Minerals							350	1-5	(7)

b) Tailings effluents

Liquid releases from tailings areas consist of the main effluent overflow and seepage through dykes. In Canada, the current practice for operating tailings area is to add a barium chloride (BaCl_2) solution to the clear decant from the tailings pond and the collected seepage at a rate of about 10 ppm and allow the precipitated radium-barium sulphate to settle in a lagoon. Flocculents may be added to improve settling. Further barium chloride addition may be made to the settling pond effluent to improve the removal of radioactivity. This passes through a second settling pond prior to discharge to natural watercourses. Dissolved radium removal is over 99% efficient; typically, claimed reduction in soluble radium-226 is from an initial level of 300 to 1000 pCi/l in the tailings decant, to a level of 2 to 10 pCi/l in the settling basin decant (Table 8).

Although it is generally accepted that addition of barium chloride to tailings pond effluent mill, if operated properly, reduce soluble radium-226 to below 10 pCi/l two potential difficulties remain: assurance that the precipitated barium sulphate is not discharged as a solid to natural watercourses and the question of long term stability of impounded barium radium sulphate.

Contamination of receiving waters by suspended barium radium sulphate will result in an increase in the radium content of river sediments. Subsequent leaching of this sediment by river water may result in drinking water contamination downstream from the mill. This has resulted in the AECB setting a target level of 10 pCi/l total radium-226 in effluents discharged to natural watercourses. Both of the above concerns have been addressed in the current literature. Reviewing published data, Itzkovitch [] concluded that:

- Soluble Th-230 is adequately controlled in acid mill effluents by neutralization to pH 7 or higher. No added treatment for Th-230 control is required for carbonate mill effluent.
- Of the options evaluated on pilot scale or employed in uranium mills for Ra-226 control, precipitation of barium radium sulfate by controlled additions of barium chloride is the "Best Available Practical Technology".
- Levels of less than 3 pCi/l soluble radium will be difficult if not impossible to achieve using barium chloride. A lower limit of 10 pCi/l would appear realistic.
- The target level of 10 pCi/l total radium in tailings effluent will not be attained using simple settling of barium radium sulfate. Use of flocculants appears to have promise.
- Redissolution of precipitated barium radium sulfate poses a potential long-term environmental problem. Many of the factors affecting the kinetics of redissolution need to be extensively studied.
- Alternatives to barium radium precipitation as a means of controlling radium need to be developed if the provincial (Ontario) standard of 3 pCi/l soluble and the AECB target of 10 pCi/l are to be met consistently.

A typical analysis of radioactive concentrations in the last pond water prior to discharge into water courses is presented in Table 9 [33].

TABLE 9 - TYPICAL TREATED TAILINGS POND EFFLUENT CONCENTRATIONS
AT ELLIOT LAKE AREA (DISSOLVED VALUES)

Radionuclides	Concentration (pCi/litre)
Ra-226	1 - 7
U-238, U-234	10 - 20
Th-230	1 - 20
Pb-210	<4
Po-210	5 - 30
Th-232	1 - 5
Th-228	1 - 3
Ra-228	<2

Liquid effluent flow from non-operating tailings ponds and areas are generally smaller than that at operating tailings sites. However effluent treatment at non-operating sites may not be as effectively carried out as at operating tailings sites.

c) Seepage

Seepage refers to the movement of liquid from a tailings area to the adjacent surroundings. Such movement may be through containment dams and dykes, foundation material, or be superficial.

The porosity of tailings permits seepage in both the lateral and vertical planes. Most tailings impoundment dams and dykes constructed in the Elliot Lake region during the early years of operation consisted of mine waste rock and local gravel, allowing for the passage of greater quantities of seepage than more recent structures.

Seepage characteristics, based on several chemical analyses of untreated seepages in the Elliot Lake area, have shown the following ranges : dissolved Ra-226 (5-42 pCi/l), total Thorium (1000-10000 pCi/l) and dissolved Lead-210 (30-1000 pCi/l) [37].

4.7 Tailings

The processing of uranium ore results in the need to dispose of large quantities of solid wastes and liquid effluents. With the exception of the recovered uranium and some process losses, tailings account for practically all of the ore solids and the process additives, including water. It is estimated that uranium tailings from mining operations in the Elliot Lake area amount to about 80×10^6 tons covering more than 400 hectares (1000 acres) [38]. Typical activities delivered to tailings areas are shown in Table 10. []

TABLE 10 - RADIOACTIVITY DELIVERED TO TAILINGS AREAS

Tailings Area	Tailings Mass 10 ⁶ Metric Tons (as of Jan '76)	NATURAL URANIUM CHAIN										THORIUM CHAIN	
		Uranium 238		Thorium 230		Radium 226		Lead 210		Thorium 232			
		Mass (Kg)	Activity*	Mass (Kg)	Activity*	Mass (Kg)	Activity*	Mass (Kg)	Activity*	Mass (Kg)	Activity*	Mass (Kg)	Activity*
Denison	23	1.61 x 10 ⁶	536	552	10,695	10.8	10,695	1.31 x 10 ⁻¹	10,695	6.90 x 10 ⁶	759		
Lacnor	2.7	1.89 x 10 ⁵	62.9	64.8	1,256	1.27	1,256	1.54 x 10 ⁻²	1,256	8.10 x 10 ⁵	89.1		
Mordic	12	8.40 x 10 ⁵	280	288	5,580	5.64	5,580	6.84 x 10 ⁻²	5,580	3.60 x 10 ⁶	396		
Panel	3.3	2.31 x 10 ⁵	76.9	79.2	1,535	1.55	1,535	1.88 x 10 ⁻²	1,535	9.90 x 10 ⁵	109		
Pronto	2.1	1.47 x 10 ⁵	48.9	50.4	977	0.987	977	1.20 x 10 ⁻²	977	6.30 x 10 ⁵	69.3		
Quirke	14.75	1.03 x 10 ⁶	344	354	6,859	6.93	6,859	8.41 x 10 ⁻²	6,859	4.43 x 10 ⁶	487		
Sp. American	0.45	3.15 x 10 ⁴	10.5	10.8	209	0.212	209	2.57 x 10 ⁻³	209	1.35 x 10 ⁵	14.9		
Stanleigh & Milliken	6.8	4.76 x 10 ⁵	158	163	3,162	3.20	3,162	3.88 x 10 ⁻²	3,162	2.04 x 10 ⁶	224		
Stanrock & Can-Met	5.5	3.85 x 10 ⁵	128	132	2,558	2.59	2,558	3.14 x 10 ⁻²	2,558	1.65 x 10 ⁶	182		

*Activity expressed in curies



Mill tailings contain most of the radioactive decay products of uranium, a little residual uranium (about 10%) and Th-232 and its decay products. Uranium tailings contain about 85% of the initial radioactivity of the source ore. It is estimated that 1 tonne of tailings has a radioactivity of at least 5 millicuries [33]. About 90% of this activity is due to the uranium-238 series. The most troublesome daughter isotopes of uranium-238 have been identified as thorium-230, radium-226 and lead-210.

With no parent remaining, the thorium-234 and protactinium-234, which are relatively short-lived radionuclides, decay out of the mill wastes. Long-lived daughters such as Ra-226, Th-230, Pb-210 and Po-210, which are expected to be found in equal amounts in tailings since uranium and its daughters are in secular equilibrium in the ore, have shown significant concentration differences. Some variation is expected since segregation may result from chemical processes in the mill and differences in the leach characteristics of radionuclides. Because of segregation and differences in the physical transport of sands and slimes in the tailings pond, varying levels of radioactivity are expected over the surface of tailings and it is difficult to obtain a representative sample.

A typical radiological tailings composition is shown in Table 11 [33].

4.8 Gaseous Effluents

4.8.1 Introduction

Radioactive gaseous effluents from uranium mining and milling activities consist essentially of radon which originates mainly from the U-238 decay series (Rn-222) and to a lesser extent from the Th-232 decay series (Rn-220 or thoron). Although it is an inert gas, radon is slightly soluble in water.



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- 4.31 -

TABLE 11. TYPICAL COMPOSITION OF ELLIOT LAKE AREA TAILINGS
(All values in pCi/gm)

	Ra-226	Tn-230	Th-232	Th-228
<u>Operating Tailings</u>				
Long Lake	70	80	15	15
Quirke	250	90	10	10
<u>Non-Operating Tailings</u>				
Nordic	240	30	5	10
Standock	110	30	10	30

Thoron (Rn-220) is considered to have a negligible environmental impact comparatively to Rn-222, owing to its short half-life (54 sec.) which limits its diffusion through a material and to the low contribution of the thorium-232 series to the total activity of the ore (Table 2).

Radon (Rn-222), being an unreactive gas with a 3.82 decay half-life, may escape from its immediate environment, whether that environment is an ore body, a tailings pile, or any material which contains radium-226, its radioactive parent. Whenever in presence of its radioactive parent, radon is continuously produced by the decay of radium-226. When radon undergoes alpha decay, there follows within a short time a series of four more decays (two by alpha emission) before relatively stable Pb-210 (half-life of 21 years) is reached. The longest half-life among the intervening nuclides is only 27 minutes. The radon daughters are not gases, but heavy metals, and rapidly become attached to available surfaces, usually to small airborne particles or aerosols.

Radon atoms are created within the material in which the radium parent is present. If the transformation occurs deep within a large particle of material, it is most unlikely that the energy of the radon atom will cause it to leave the particle and enter the spaces between the particles of the material. However, when the particles are very small, the probability of this escape is high.

The escape of radon atoms to this inter-particle space is referred to as emanation. Having emanated they are then free to migrate anywhere in the inter particle space (i.e the gas or water in the soil, tailings or rock). It is assumed that only about 20% of the radon produced is available for release or emanation from the mineral grains in which it is produced.

The radon which has emanated to the inter-particle spaces migrates from one location to another by diffusion or transport. Diffusion causes radon to travel from regions of higher concentration to regions of lower concentration. The flow rate is determined by the physical characteristics of the diffusion medium and by the concentration gradient. The overall measure of the physical characteristics is a factor known as the diffusion coefficient which can be determined experimentally. Generally, the diffusion coefficient, and thus the flow rate, will be lower for liquid and solid materials than for gases.

In transport, radon is carried by water or air from one location to another. Even with very low water or air velocities, the migration by transport will be much higher than would occur by diffusion. Migration by both mechanisms may occur simultaneously.

When radon enters the ambient air environment it is said to have been exhaled. Thus one may speak of an exhalation rate or flux from the surface of an ore body, a tailings area or a body of water.

The rate of exhalation depends upon both the emanation rate and migration rate. When the emanation factor is small, little of the radon produced is able to migrate. If the migration is slow because of the diffusion coefficient and transport velocity are low, less of the radon will be exhaled because it will decay before it reaches the air/material interface.

4.7.2 Radon emission sources

Airborne radon and radon daughters releases to the environment have been reported to originate primarily from the following sources:

- ° mine ventilation exhaust
- ° ore pile emanation
- ° mill ventilation exhaust
- ° tailings emanation
- ° emanation from dispersed radium-bearing materials.

a) Mine ventilation

Radon-222 is reported to be the main radioactive effluent in the ventilation air from underground as well as open-pit mines. Typical radon emission rates from two underground mines in the Elliot Lake region, as reported by McLaren [37], are about 10-13 $\mu\text{Ci/sec.}$ per 1000 metric tons of ore ($0.1\% \text{ U}_3\text{O}_8$) per day. Radon releases from mine ventilation exhaust at the Beaverlodge mine in Saskatchewan are of the order of 50-100 $\mu\text{Ci/sec}$ [39].

Because of the diffuse nature of the open-pit mining operations, no direct measurement of radon-222 total emission can be made; emissions rates of radon from open-pit mines are currently estimated using calculational methods.

Radon emission rate from open-pit operations (500 tons of ore per day) at Cluff Lake's "D" ore body, where a uranium content up to 30% is found, has been calculated to be about 11 microcuries per second [39]. The emission is estimated to be relatively low in spite of the high ore uranium content due to the use of an inert sand shield over the mined area of the pit.

b) Ore pile emanation

During the period of storage on the ore pad and bins prior to processing in the mill, radioactive decay of radium-226 in the ore produces radon continuously. Total quantity of radon released depends on the emanating power (i.e. fraction of radon released), the storage time and the concentration of radium-226 in the ore. A conservative approach is to assume that all of the radon available for release (20%) actually escapes from the ore pile as soon as it forms.

Contribution from ore storage pad and bins for a 1800 metric tons per day mill capacity, assuming a 12-day period of storage and 0.1% U content in ore, was estimated to be about 68 Ci/year⁽¹³⁾. Calculated emissions from the high-grade ore stock pile at Cluff Lake, Saskatchewan, is reported to be 3 μ Ci/sec (95 Ci/year)[39].

c) Mill ventilation exhaust

Radon emissions from the Denison and Quirke mills have been reported as 5-12 μ Ci/sec. with the main sources being the leaching pachuchas and the crushing and grinding circuits. Radon releases from Cluff Lake's mill is estimated to be about 2 μ Ci/sec. [39]. Radon releases from mills are thus about one order of magnitude lower than from mines.

d) Tailings emanation

Radon is continually generated in tailings piles where it diffuses within tailings materials and escapes to the atmosphere. Radon-222 gas will emanate from the tailings pile for thousands of years unless both the radium-226 parent (half-life, 1620 years) and the thorium-230 grandparent (half-life, 83,000 years) are removed or a radon diffusion barrier is placed over the pile.

The rate of radon exhalation at the interface between a tailings pile and the atmosphere is dependent on (1) the rate of generation of radon from the parent radium, (2) the concentration of radon in the soil pore space, (3) diffusion through the pore spaces, and (4) transport phenomena.

The emanation rate of radon from tailings has equally been reported to be strongly influenced by the moisture content, the particle size and texture (sands or slimes) of tailings, temperature, pressure, atmospheric stability and weather conditions. Snow cover, rainfall and weather conditions may change the emanation rate by a factor of 10 or more. Calculations have shown that complete saturation of dry tailings with water may reduce its radon emissions by a factor of about 25.

Radon diffusion rates from tailings area in Canada are reported to be reduced during approximately 45% of the year based on snow cover, frost and rainfall data. Experiments have shown that 15-25% of the radon produced escapes from the tailings grains and that effectively only the radon produced in the top 1 meter of tailings will diffuse to the atmosphere.

Radon emanation rates from Elliot Lake area tailings measured during the summer typically ranged from 10 pCi/m²-sec. to 500 pCi/m²-sec. Levels of ambient radon concentrations over operating and non-operating tailings areas ranged from less than 2 pCi/l to nearly 10 pCi/l. Off the tailings, levels are reported to be less than 1 pCi/l within 90 meters of the tailings areas and in most instances to be near the detection limit of the equipment[37].

e) Emanation from dispersed radium-bearing materials

During operation of the mill, radium-bearing ore and tailings dusts are released and settle onto the ground surfaces around the mill site. The radium that settles out will continue to undergo radioactive decay to produce a secondary source of radon gas. The release of radium reportedly comes almost all from dusting of exposed tailings surfaces.

The release of particulate material from tailings areas is difficult to quantify, being dependent on the moisture content of the tailings material, its particle size distribution, and on wind velocity. However, radon emanation from this source is not thought to constitute a significant contribution to the total radon production.

4.9 Health Hazards Due to Radon

When converting Radon concentration from picocuries per liter to working levels (WL), the following results were obtained in and around Elliot Lake [40] :

TABLE 12. Radon Daughter Concentrations
In and Around Elliot Lake, in W.L.

<u>Location</u>	<u>Average Concentration,</u> <u>in W.L.</u>
1. About 60 metres between two mine exhaust shafts	0.0660
2. 300 metres from exhaust shaft	0.0080
3. 300 metres from exhaust shaft	0.0012
4. 1500 metres from exhaust shaft	0.0010
5. 100 metres from an operating tailings facility	0.0028
6. A townsite 600 m from dormant tailings facility	0.0024
7. A townsite between exhausts shafts, mill exhaust and tailings, minimum distance 1000 metres	0.0071
8. Elliot Lake Municipal Building - no evident source of radon daughters within 200 metres	0.0008

N.B.- Natural concentration is around 0.001.

The radiological risk resulting from an exposure of 1 WL is still controversial, but can be taken approximately as 3×10^{-4} fatal lung cancers.

4.10 Collective Doses

The Nuclear Energy Agency of the OECB has organized a study of the applicability of the ICRP system of dose limitation to the management of uranium mill tailing. For this study, three reference sites have been defined, one in Canada, one in the USA and one in Australia. First results of this study have been recently presented [28]. For Canada, the valey dam impoundment is the base case, with several other options included, such as radium-thorium removal from the tailings.

Water transport of radionuclides and atmospheric dispersion of radon have been modelled. Waterborne contaminants, resulting from all leakable radioactive elements, would produce a collective dose commitment (estimated for 10^4 years; the plateau is reached in about 500 years) of about 2000 person-Sieverts. The collective dose commitment produced by radon is 3 orders of magnitude lower. The collective dose commitments calculated are delivered at individual dose rates from a few tenths of microsieverts per year (natural background radiation dose is 1 millisievert per year) down many orders of magnitude. From [28] is also possible to infer that if the collective doses are only summed for the population group when the individual dose rates average more than a microsievert per year (trivial dose ?), then the collective dose commitment would be an order of magnitude lower. The health hazards would then be around two additional fatal cancers in 10 000 years.



CHAPTER 5. REFINING AND FUEL FABRICATION

Refining and fuel fabrication activities are very low risk operations as compared with those resulting from other activities within the nuclear. As compared with mining and milling, much smaller quantities of material have to be handled (about 1/1 000); the only radioactive material present, except as traces, is uranium. Process losses are very small and control is easy. As compared with the fuel in the reactor and the spent fuel, the natural uranium has a very low specific activity.

There are few actual measurements on individual doses to members of the general population resulting from normal operation of refining and fuel fabrication installations. According to [36] the highest culculated dose exposures are less than 3×10^{-5} Sv/year from refining and less 1×10^{-5} Sv/year from fuel fabrication (as compared with less than 20×10^{-5} Sv/year from mining and milling operations).

Collective lifetime dose commitments in U.S. for refining and fuel fabrication have been estimated to be around 0.1 and 0.06 person-sieverts, respectively (vs 6, 1.2 and 0.76 person-sieverts corresponding to mining, milling and reactor operation).

Accidental releases of radioactive materials are minimized through good engineering practices and strict AECB requirements.

Transportation hazards of fresh fuel is negligible with respect to those resulting from spent fuel (discussed in the Chapter).

CHAPTER 6. NUCLEAR POWER PLANT OPERATION

6.1 Introduction

No single other activity within the entire fuel cycle has been more thoroughly studied than the operation of a nuclear power generating station. The volume of documentation on reactor safety-related aspects is staggering and few people could pretend to be actually knowledgeable on all these aspects and to summarize them properly. Therefore, studies presented by research and development organizations (such as AECL in Canada), utilities (like Ontario Hydro), regulatory bodies (like AECB), parliamentary commissions (like the Porter Commission) and international bodies (such as ICRP, UNSCEAR, etc), are thoroughly scrutinized for accuracy before being publicly issued. All these studies point to the same direction: with proper care, nuclear power plants are safe to operate with minimal radiological impact on the population.

Available information on public health hazards from operation of a CANDU nuclear power plant will now be reviewed.

6.2 Risk from Normal Operation*

6.2.1 Source terms

Releases of radioactive elements are carefully monitored by the utilities and by AECB. Chronic radioactive emissions have been, in Canada, consistently below the design target of 1% of derived emission

* Material presented in this section is taken from reference [41] and from Background Paper No. 3 [42].

limits (DEL)*. The performance record of the Pickering generating station, with respect to this target, can be seen in Figure 8 [41], for both air and water emissions. Actual releases, from all canadian nuclear installations (from 1972 to 1979) can be seen in Tables 13 and 14.

Ontario Hydro has prepared a preliminary assessment of the impact of the chronic radiological emissions from a nominal 4 x 850 MWe nuclear generating station. The source model, normalized to a production of 1 GW-a electric energy has characteristics shown in Tables 15 and 16 [43].

6.2.2 Dispersion models

First, exposure pathways to man have to be identified. This is graphically represented in Figures 9 [42], where as physical events to be modelled are shown in Figures 10 and 11 [44] for atmospheric and aquatic releases respectively.

In the Ontario Hydro study, an american [45] and a canadian model [46] have been used**. Results from this study are shown in Table 17 (in rems/year) these estimates have to be compared with global results, based on measurements, for Pickering GSA and for Bruce Nuclear Power Development (Table 18).

*The DEL are calculated releases which would result at the most, in the maximum permissible doses for members of the public; the computations are made using conservative dispersion models.

**In Europe, the Commission of the European Communities has developed its own model [44]. In U.S.A. U.S. Nuclear Regulatory guides (1.109, 1.111, 1.112 and 1.113) are followed for computing annual doses to man from routine radioactive releases.

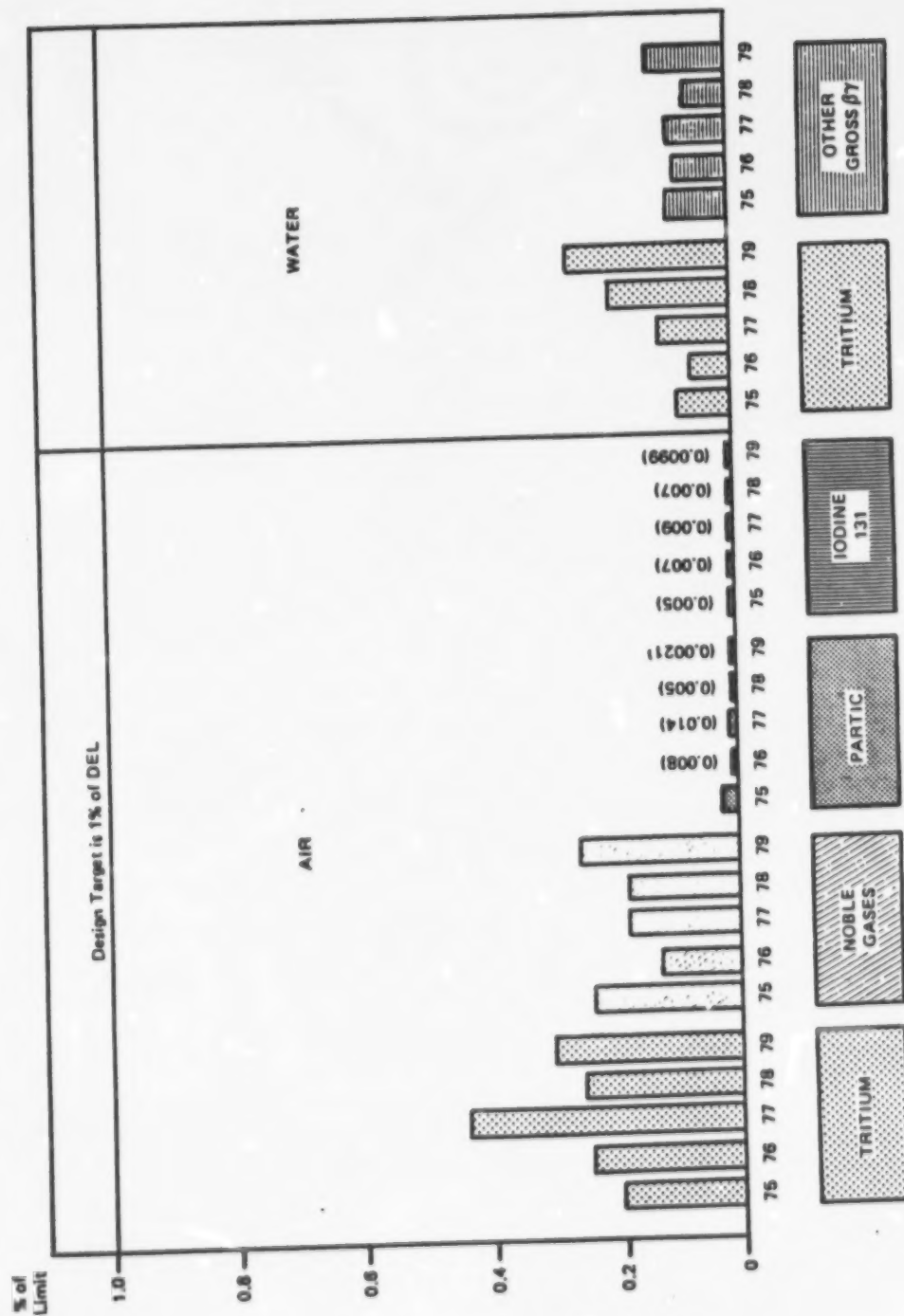


FIGURE 8
Radioactive Emissions - Pickering GS A

	1972 ⁺ Ci/day	1973 ⁺ Ci/day	1974 ⁺ Ci/day	1975 ⁺ Ci/day	1976 ⁺ Ci/day	1977 ⁺ Ci/day	1979 ⁺ Ci/day
NRU							
³⁹ Ar	-	4 500	5 000-10 000*	-	-	-	-
Tritium (oxide)	44**	4.2	6.6	9.6	9.6	19.3	-
¹³¹ I	<0.01***	<0.002	<0.004	2.2·10 ⁻²	2.7·10 ⁻³	1.1·10 ⁻²	-
¹²⁵ I	-	-	-	2·10 ⁻²	4.9·10 ⁻³	8.2·10 ⁻²	-
Particulates	-	-	-	6·10 ⁻⁶	1.3·10 ⁻⁵	1.0·10 ⁻⁵	-
WRE							
Ar	-	28	34	-	-	-	-
Noble fission gases	-	0.6	0.4	-	-	-	-
¹³¹ I	<<0.001	<<0.001	<<0.001	1.8·10 ⁻⁵	5·10 ⁻⁷	N.D.	-
¹²⁵ Sb	-	<3·10 ⁻⁹	<8·10 ⁻⁶	9.9·10 ⁻⁶	N.D.	N.D.	-
⁹⁵ Zn- ⁹⁵ Nb	-	<3·10 ⁻⁹	<3·10 ⁻⁴	3·10 ⁻⁷	N.D.	N.D.	-
⁷⁶ As	-	<3·10 ⁻⁹	<1·10 ⁻⁶	-	-	-	-
Tritium (oxide)	<0.25	<0.25	<0.25	<0.3	<0.3	<0.24	-
Particulates	-	-	-	3.0·10 ⁻⁵	1·10 ⁻⁴	3.04·10 ⁻⁵	-
NPD							
Tritium (oxide)	10	3.6 10 ¹	2.1 10 ¹	35	36	36.2	-

Taken from AECB-1068, 1115, 1135

The amount discharged from NRU reactor depended on the amount of CO₂ being injected into the annulus to restrict corrosion.

Most of this was due to an accidental release, without which the average daily release would have been 16 Ci/day.

Radioiodines were not measured routinely in 1972, but spot samples indicated <0.01 Ci/day, which was typical of the results obtained routinely from 1964 to 1970.

TABLE 13 (Cont'd) Average Atmospheric Release of Radioactive Material from Nuclear Reactors, 1972-1979

	1972 ⁺ Ci/day	1973 ⁺ Ci/day	1974 ⁺ Ci/day	1975 ⁺ Ci/day	1976 ⁺ Ci/day	1977 ⁺ Ci/day	1979 ⁺ Ci/day
Noble Gases⁺⁺	-	3.3·10 ⁻²	3·10 ⁻²	-	-	-	-
¹³¹ I	<5·10 ⁻⁶	1·10 ⁻⁴ +++	3·10 ⁻⁶	N.D.	N.D.	N.D.	-
Particulates	-	3·10 ⁻⁵ +++	N.D.	3.7·10 ⁻⁷	N.D.	N.D.	-
<u>Douglas Point</u>							
Tritium (oxide)	20	5.5·10 ⁻¹	1.4·10 ⁻¹	41	94	32.3	17.4
Noble Gases⁺⁺	-	3.7·10 ⁻²	5.8·10 ⁻¹	-	-	-	5.01·10 ⁻¹
¹³¹ I	<6·10 ⁻⁸	5.3·10 ⁻⁵	1.3·10 ⁻⁴	9.9·10 ⁻⁵	4.4·10 ⁻⁵	2.52·10 ⁻⁶	3.07·10 ⁻⁶
Particulates	-	1.8·10 ⁻⁶	1.3·10 ⁻⁶	9.9·10 ⁻⁷	6.2·10 ⁻⁷	0.42·10 ⁻⁶	5.9·10 ⁻⁷
<u>Pickering A</u>							
Tritium (oxide)	39	1·10 ⁻²	6.8·10 ⁻¹	55	66	120.5	83.55
Noble Gases⁺⁺	-	1.1·10 ⁻²	1.2·10 ⁻¹	-	-	-	15.9
¹³¹ I	3.6·10 ⁻⁴	1.5·10 ⁻⁵	1.1·10 ⁻⁵	2.6·10 ⁻⁶	4.1·10 ⁻⁶	0.52·10 ⁻⁵	5.66·10 ⁻⁶
Particulates	-	3·10 ⁻⁵	9.3·10 ⁻⁵	2.0·10 ⁻⁵	1.1·10 ⁻⁵	1.97·10 ⁻⁵	2.96·10 ⁻⁶
<u>Gentilly</u>							
Tritium (oxide)	-	.14	7	5	3	2.6	-
Noble Gases⁺⁺	19	<10	<10	-	-	-	-
¹³¹ I	0.002	<2·10 ⁻⁶	<2·10 ⁻⁶	<2·10 ⁻⁶	<2·10 ⁻⁶	1·10 ⁻⁶	-
Particulates	-	<5·10 ⁻⁷	<5·10 ⁻⁷	<5·10 ⁻⁷	<5·10 ⁻⁷	<5·10 ⁻⁷	-

⁺⁺ Ci/MeV

+++ Value given in based on the detection limit of stock effluent monitoring equipment

TABLE 13 (cont'd) Average Atmospheric Releases of Radioactive Effluents from Canadian Nuclear Installations in 1972-1979

	1972 ⁺ Ci/day	1973 ⁺ Ci/day	1974 ⁺ Ci/day	1975 ⁺ Ci/day	1976 ⁺ Ci/day	1977 ⁺ Ci/day	1979 ⁺ Ci/day
<u>Bruce A</u>							
Tritium (oxide)	-	-	-	-	<2.5·10 ⁻⁴	23.3	101.84
¹³¹ I	-	-	-	-	<5·10 ⁻⁷	3.4·10 ⁻⁵	5.66·10 ⁻⁶
Particulates	-	-	-	-	<1.4·10 ⁻⁷	0.7·10 ⁻⁵	9.11·10 ⁻⁶
Noble Gases ⁺⁺	-	-	-	-	-	-	224.68

⁺⁺ Ci/Mev

TABLE 14 Liquid Waste Discharge from Canadian Nuclear Installations in 1972-1979

CNRL	1972 Ci/day	1973 Ci/day	1974 Ci/day	1975 Ci/day	1976 Ci/day	1977 Ci/day	1979 [*] Ci/day
Gross- β - γ	-	-	-	-	-	$4.3 \cdot 10^{-1}$	-
Gross- β *	$8.3 \cdot 10^{-2}$	$14.0 \cdot 10^{-2}$	$12.0 \cdot 10^{-2}$	$1.2 \cdot 10^{-1}$	$5.0 \cdot 10^{-1}$	-	-
Tritium (oxide)	8.6	7.4	12.6	9.4	8.3	7.0	-
^{239}Pu	$9.9 \cdot 10^{-6}$	$17 \cdot 10^{-6}$	$30 \cdot 10^{-6}$	$2.7 \cdot 10^{-5}$	$1.3 \cdot 10^{-5}$	$9.4 \cdot 10^{-6}$	-
^{239}Np	-	$4.59 \cdot 10^{-2}$	$17.0 \cdot 10^{-2}$	-	-	$6.6 \cdot 10^{-2}$	-
^{90}Sr	$2.5 \cdot 10^{-3}$	$3.5 \cdot 10^{-3}$	$4.3 \cdot 10^{-3}$	$5.0 \cdot 10^{-3}$	$5.8 \cdot 10^{-3}$	$8.5 \cdot 10^{-3}$	-
^{137}Cs	$1.3 \cdot 10^{-3}$	$4.1 \cdot 10^{-3}$	$4.1 \cdot 10^{-3}$	$5.4 \cdot 10^{-3}$	$7.6 \cdot 10^{-3}$	$11.3 \cdot 10^{-3}$	-
^{134}Cs	-	$2.4 \cdot 10^{-3}$	$2.1 \cdot 10^{-3}$	$2.0 \cdot 10^{-2}$	$4.2 \cdot 10^{-2}$	$4.8 \cdot 10^{-2}$	-
^{106}Ru	$2 \cdot 10^{-4}$	$1 \cdot 10^{-4}$	$2 \cdot 10^{-4}$	$2.3 \cdot 10^{-3}$	$2.6 \cdot 10^{-3}$	$2.5 \cdot 10^{-3}$	-
^{138}Ba	-	$1.36 \cdot 10^{-2}$	$1.01 \cdot 10^{-2}$	$3.3 \cdot 10^{-2}$	$6.2 \cdot 10^{-2}$	$7.1 \cdot 10^{-2}$	-
^{131}I	-	$1.35 \cdot 10^{-2}$	$1.41 \cdot 10^{-2}$	$2.4 \cdot 10^{-2}$	$4.6 \cdot 10^{-2}$	$4.8 \cdot 10^{-2}$	-
^{95}Zr	$1.9 \cdot 10^{-3}$	$13.8 \cdot 10^{-3}$	$7.4 \cdot 10^{-3}$	$2.8 \cdot 10^{-2}$	$5.3 \cdot 10^{-2}$	$5.6 \cdot 10^{-2}$	-
^{94}Mn	$2.8 \cdot 10^{-3}$	$6.6 \cdot 10^{-3}$	$11.0 \cdot 10^{-3}$	$5.9 \cdot 10^{-3}$	$7.7 \cdot 10^{-3}$	$9.3 \cdot 10^{-3}$	-
^{65}Sc	$5.2 \cdot 10^{-3}$	$11.2 \cdot 10^{-3}$	$17.5 \cdot 10^{-3}$	$1.3 \cdot 10^{-2}$	$1.3 \cdot 10^{-2}$	$1.5 \cdot 10^{-2}$	-
^{59}Fe	$3.1 \cdot 10^{-3}$	$4.5 \cdot 10^{-3}$	$8.6 \cdot 10^{-3}$	$6.9 \cdot 10^{-3}$	$8.5 \cdot 10^{-3}$	$1.2 \cdot 10^{-2}$	-
^{65}Zn	$7.2 \cdot 10^{-3}$	$13.9 \cdot 10^{-3}$	$24.2 \cdot 10^{-3}$	$3.3 \cdot 10^{-2}$	$2.3 \cdot 10^{-2}$	$1.5 \cdot 10^{-2}$	-
^{60}Co	-	$3.7 \cdot 10^{-3}$	$3.3 \cdot 10^{-3}$	$1.5 \cdot 10^{-3}$	$4.8 \cdot 10^{-3}$	$9.6 \cdot 10^{-3}$	-
WRE (WR-1)	-	<0.001	<0.001	$1.5 \cdot 10^{-2}$	$1.6 \cdot 10^{-1}$	-	-
Gross- β	-	-	-	-	-	-	-
Gross- β - γ	-	-	-	-	-	0.25	-

* Measured five days after release.

+ Taken from AECB-1068, 1115, 1135

++ Combined releases from WR-1 and Hot Cells Facility. Note the WRE data given in the earlier report AECB-1068 (for years 1972, 1973 and 1974) applies to WR-1 releases, except for ^{95}Zr - ^{95}Nb which is released from the Hot Cells Facility only. ^{95}Zr - ^{95}Nb releases in 1974 should read as $1.1 \cdot 10^{-6}$ Ci/day* instead of $3 \cdot 10^{-5}$ Ci/day*.

TABLE 14 (cont'd) Liquid Waste Discharge from Canadian Nuclear Installations in 1972-1979

	1972 Ci/day	1973 Ci/day	1974 Ci/day	1975 Ci/day	1976 Ci/day	1977 Ci/day	1979 Ci/day
Alum (oxide)	2.0	13.2	2.6	6.1	4.1	2.7	-
β-β	$0.7 \cdot 10^{-4}$	$2.7 \cdot 10^{-4}$	$2.7 \cdot 10^{-4}$	$3.5 \cdot 10^{-4}$	$8.9 \cdot 10^{-5}$	-	-
β-β-γ	-	-	-	-	-	$10.4 \cdot 10^{-5}$	-
Gas Point							
Alum (oxide)	12.0	21.4	9.4	7.4	12.8	5.4	9.3
β-β	$11 \cdot 10^{-3}$	$5.5 \cdot 10^{-3}$	$3.4 \cdot 10^{-3}$	$2.0 \cdot 10^{-3}$	$1.9 \cdot 10^{-3}$	-	-
β-β-γ	-	-	-	-	-	$0.6 \cdot 10^{-3}$	$6.7 \cdot 10^{-4}$
Spring A							
Alum (oxide)	2.7	14.1	40.0	29	16.6	52.1	$1.03 \cdot 10^2$
β-β	$7.6 \cdot 10^{-3}$	$4.9 \cdot 10^{-3}$	$7.2 \cdot 10^{-3}$	$2.6 \cdot 10^{-3}$	$2.2 \cdot 10^{-3}$	$2.2 \cdot 10^{-3}$	-
β-β-γ	-	-	-	-	-	-	$2.7 \cdot 10^{-3}$
β A							
Alum (oxide)	-	-	-	-	$<1.2 \cdot 10^{-1}$	2.65	39.83
β-β	-	-	-	-	$<6.0 \cdot 10^{-6}$	-	-
β-β-γ	-	-	-	-	-	$1.75 \cdot 10^{-3}$	$1.54 \cdot 10^{-2}$

The results are averaged over a period of 5 months starting July, 1976. (Bruce A Unit 2 and 1 started operation in July, 1976 and December, 1976, respectively. Units 3 and 4 are under construction).

- 6.8 -

TABLE 14 (cont'd) Liquid Waste Discharge from Canadian Nuclear Installations in 1972-1979

	1972	1973	1974	1975	1976	1977	1979
	Ci/day	Ci/day	Ci/day	Ci/day	Ci/day	Ci/day	Ci/day
ally	2.0	13.2	2.6	1	19	97	-
um (oxide)	0.150**	0.009	0.003	$4 \cdot 10^{-3}$	$4.6 \cdot 10^{-4}$	-	-
-β	-	-	-	-	-	$1.7 \cdot 10^{-3}$	-
-β-γ	-	-	-	$5.9 \cdot 10^{-3}$	$7.8 \cdot 10^{-4}$	$1.7 \cdot 10^{-3}$	-
	-	-	-	$2.8 \cdot 10^{-4}$	N.D.	$1.7 \cdot 10^{-5}$	-
	-	-	-	$1.7 \cdot 10^{-5}$	N.D.	$6.2 \cdot 10^{-5}$	-
	-	-	-	$1.1 \cdot 10^{-5}$	N.D.	$1.4 \cdot 10^{-4}$	-
	-	-	-	$5.0 \cdot 10^{-5}$	N.D.	$3.3 \cdot 10^{-4}$	-
	-	-	-	$7.1 \cdot 10^{-4}$	N.D.	$2.4 \cdot 10^{-4}$	-
	-	-	-	$3.2 \cdot 10^{-6}$	$5.0 \cdot 10^{-5}$	$5.9 \cdot 10^{-4}$	-
	-	-	-	$4.9 \cdot 10^{-6}$	N.D.	N.D.	-
	-	-	-	$3.8 \cdot 10^{-5}$	N.D.	$4.7 \cdot 10^{-5}$	-
	-	-	-	N.D.	$3.6 \cdot 10^{-8}$	$2.4 \cdot 10^{-4}$	-
	-	-	-	-	-	$5.2 \cdot 10^{-8}$	-
	-	-	-	-	-	$0.9 \cdot 10^{-4}$	-
	-	-	-	-	-	$3.7 \cdot 10^{-5}$	-
	-	-	-	-	-	$3.3 \cdot 10^{-5}$	-
	-	-	-	-	-	$3.2 \cdot 10^{-5}$	-

not typical, virtually all was released during commissioning.

Element	Emission Rate TBq/(GWe-a)
Tritium	4.63×10^2
Nobles Gases*	1.16×10^{14}
C-14	1.15
Radioactive Iodine	3.85×10^{-4}
Particles	9.66×10^{-4}

TABLE 15. Chronic Atmospheric Releases
from a Reference CANDU Reactor

* 1n (Bq-Mev)/(GWe-a)

N.B. 1 Ci/day = 13.5 TBq/year.

Element	Emission Rate Bq/(GWe-a)	Element	Emission Rat Bq/(GWe-a)
Cr-51	2.4×10^{10}	Ag-110m	3.5×10^9
Mn-54	3.2×10^7	Sb-124	6.7×10^8
Fe-59	4.4×10^8	I-131	5.6×10^9
Co-58	9.6×10^7	Cs-134	2.3×10^{10}
Co-60	8.5×10^9	Cs-136	1.5×10^9
Zn-65	3.6×10^9	Cs-137	5.6×10^{10}
Sr-89	5.6×10^9	Ba-140	5.6×10^8
Sr-90	5.6×10^8	Ce-141	1.0×10^9
Zr-95	5.9×10^9	Ce-144	1.3×10^{10}
Nb-95	4.1×10^9	Total Particle	1.8×10^{11}
Mo-99	2.1×10^{10}	Pu-239	5.6×10^7
Ru-103	1.2×10^9	Pu-240	5.6×10^7
Ru-106	2.5×10^9	H ₃	1.1×10^{15}

TABLE 16. Chronic Aquatic Releases from
a Reference CANDU Reactor

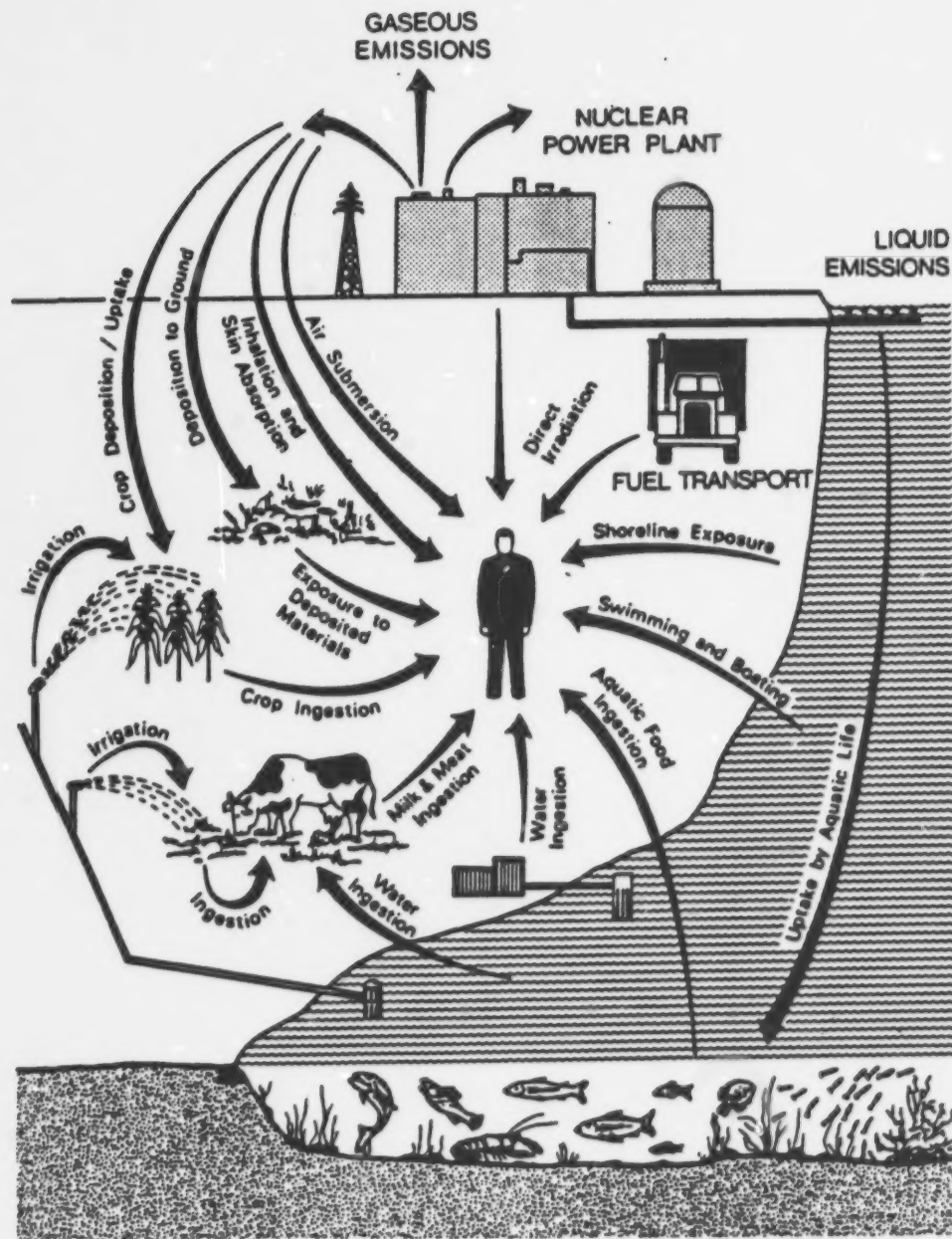


FIGURE 9
Exposure Pathways to Man

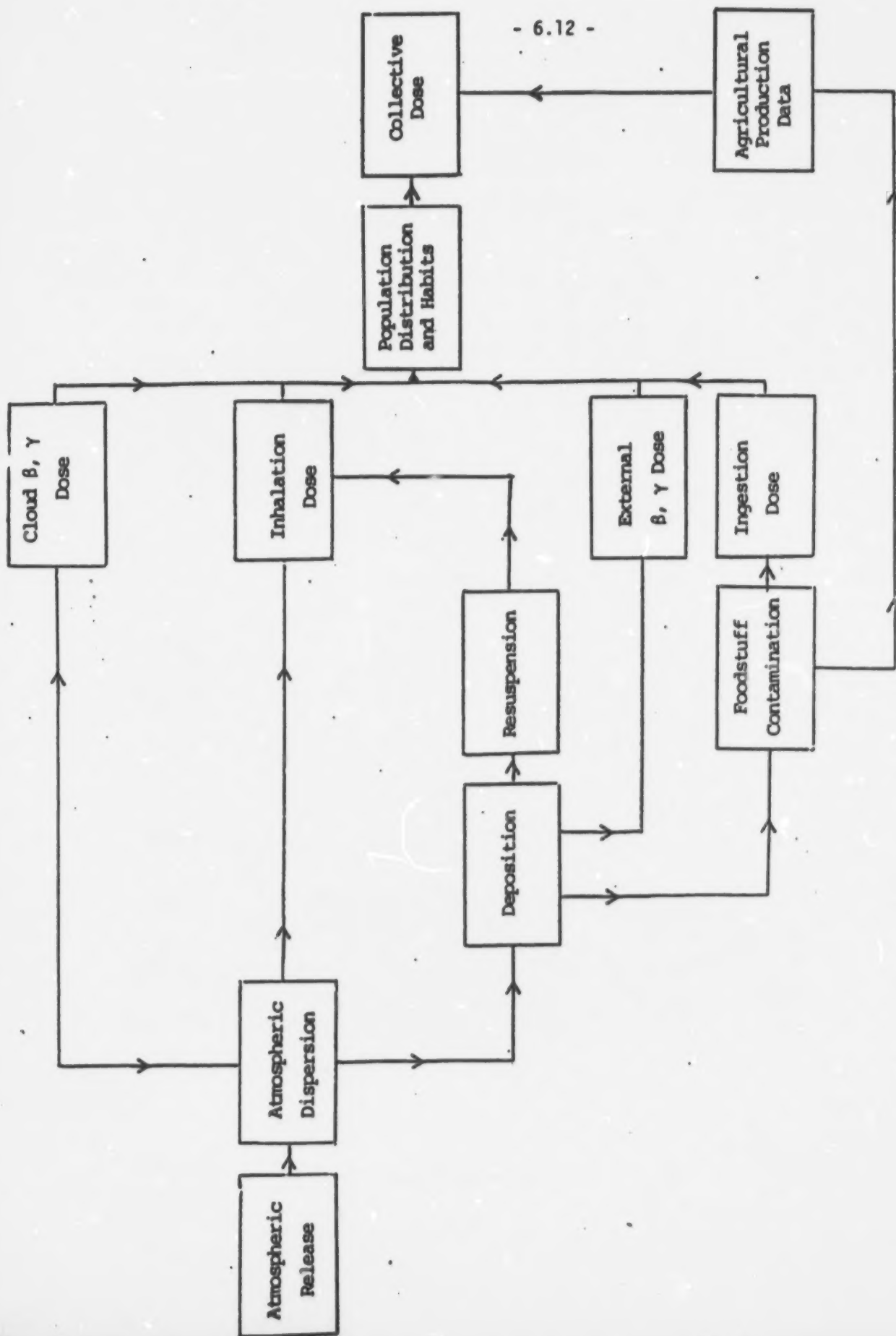


FIGURE 10 Pathways to Man Considered for Atmospheric Releases

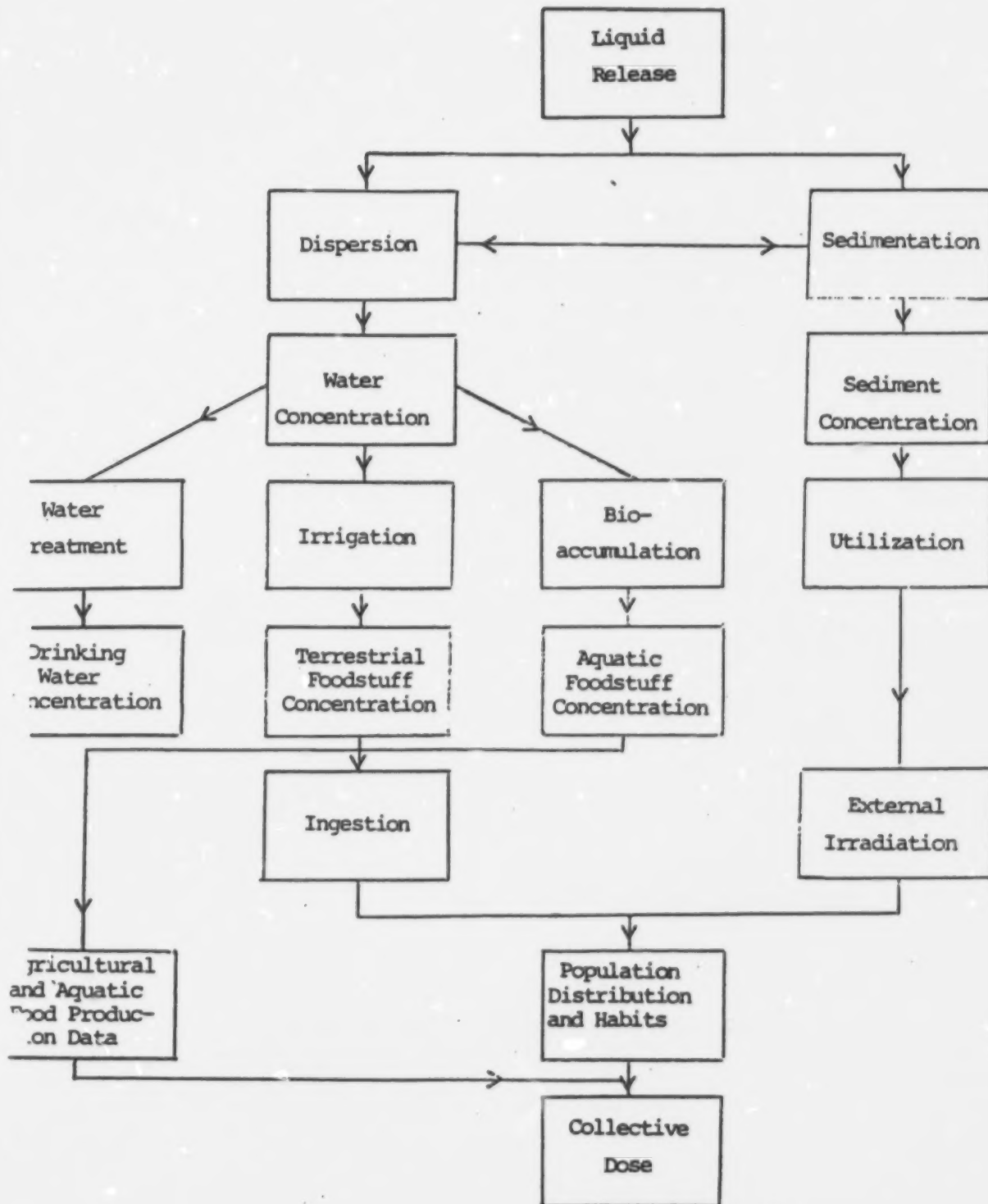


FIGURE 11 Aquatic Pathways to Man

TABLE 17
MAXIMUM BOUNDARY INDIVIDUAL DOSE ESTIMATE
AT REFERENCE 4 x 850 MWe SITE
AT MATURITY

RADIONUCLIDE OR CATEGORY	REM/A		TOTALS
	ATMOSPHERIC EMISSION PATHWAY	AQUATIC EMISSION PATHWAY	
Tritium	$< 4.3 \times 10^{-3}$	2.5×10^{-3}	$< 4.3 \times 10^{-3}$
Carbon-14	2.2×10^{-3}	—	2.2×10^{-3}
Noble Gases	1.6×10^{-3}	—	1.6×10^{-3}
Iodine — 131	0.3×10^{-3}	—	0.3×10^{-3}
Particulates	<u>0.2×10^{-3}</u>	<u>2.8×10^{-4}</u>	<u>$< 4.8 \times 10^{-4}$</u>
Totals	$< 8.6 \times 10^{-3}$	$< 3.0 \times 10^{-4}$	$< 8.9 \times 10^{-3}$

Note: Based on projected, maturity emission source terms for a reference
4 x 850 MWe station.

TABLE 18
MAXIMUM BOUNDARY INDIVIDUAL DOSE¹

	REM/A	
	ATMOSPHERIC EMISSIONS	AQUATIC EMISSIONS
PICKERING G.S. 'A'	$< 2.6 \times 10^{-3}$	$< 0.3 \times 10^{-3}$
BNPD	$< 3.6 \times 10^{-3}$	$< 0.5 \times 10^{-3}$

1 — Committed Effective Dose Equivalent

6.2.3 Collective doses

Collective dose equivalent commitments per unit emission have been calculated by a NEA group [22] and are shown in Table 19 for the long-lived isotopes H-3, C-14, Kr-85 and I-129. For a CANDU nuclear power plant, the collective dose equivalent commitments for H-3 and C-14 (the most important contributors), can therefore be estimated from Tables 15 and 19.

$$\begin{aligned}
 S_y^{10\ 000}(\text{H-3}) &= 2.1 \times 10^{-3} \frac{\text{man-rem}}{\text{Ci}} \times 3.7 \times 10^{-2} \frac{\text{Ci}}{\text{TB}_q} \\
 &\times 4.63 \times 10^2 \frac{\text{TB}_q}{\text{GW-a}} \times 10^{-2} \frac{\text{Sv}}{\text{rem}} = 3.6 \times 10^{-2} \frac{\text{man-Sv}}{\text{GW-a}} \\
 S_y^{10\ 000}(\text{C-14}) &= 2.9 \times 10^2 \frac{\text{man-rem}}{\text{Ci}} \times 3.7 \times 10^{-2} \frac{\text{Ci}}{\text{TB}_q} \\
 &\times 1.15 \frac{\text{TB}_q}{\text{GW-a}} \times 10^{-2} \frac{\text{Sv}}{\text{rem}} = 1.23 \frac{\text{man-rem}}{\text{GW-a}}
 \end{aligned}$$

6.2.4 Comparisons between models

Detailed analysis of dispersion models used in Europe, U.S.A. and Canada is provided in Background Paper No. 3 [42].

TABLE 19 Collective Dose Equivalent Commitments per Unit Emission

Nuclide	Collective dose equivalent commitments per unit emission (man-rem-Ci ⁻¹)	
	$S_{I,L,global}^{C*}$	$S_{Y,L,global}^{10000**}$
³ H	$1.7 \cdot 10^{-3}$	$2.1 \cdot 10^{-3}$
¹⁴ C	$4.0 \cdot 10^2$	$2.9 \cdot 10^2$
⁸⁵ Kr	$1.0 \cdot 10^{-3}$	$1.1 \cdot 10^{-3}$
¹²⁹ I	$1.1 \cdot 10^5$	30

* The global collective dose equivalent commitments have effective and weighted skin components.

** The global commitments have contributions from effective, weighted skin and weighted gonads doses.

6.3 Risks from Accidents

6.3.1 Introduction

The Canadian approach to nuclear power safety is described in many papers. References [47, 48, 49 and 24] are recent examples of views expressed by regulators (AECB), designers (AECL), utilities (Ontario Hydro) and independent advisers (ACNS).

All papers stress the fact that a good record of safety can only be assured by a combination of good design, good manufacturing quality control, good engineering practices during construction, commission and operation of the plant, good training and good support engineering services.

For reactors of all design, the concept of defense in depth has been extensively used in design to ensure there are several barriers between the irradiated fuel and workers and members of the public. In addition, the use of separation, independence, redundancy and diversity in the design of all safety systems and in many process systems provides the necessary insurance that the power plant can withstand several independent equipment (or system) failures. A characteristic of the CANDU design is the possibility of periodic testing of the safety system capabilities to act upon receiving trip signals, thus ensuring that the corresponding availability targets are being met.

By combining high quality equipment of low failure rates, with designed process systems of high reliability, and with independent safety systems of proven high availability, the probability of a sequence of events resulting in a large release of radioactive products has been consistently kept very low and is continuously decreasing due to the introduction of new safety measures, either imposed by the AECB or decided by the utilities on the basis of their own experience.

Safety assessments are continuously made : during the licensing process - via the preparation and the critical evaluation of safety reports -, during the commissioning of the plant and during its operation. Safety analyses have to show that the design is appropriate in meeting the AECB standards for different classes of postulated accidents, and inspection of operating records has to ensure that the actual performance meets the design specifications.

No comprehensive study such as the U.S. Reactor Safety Study ("Rasmussen Report" [29]), or the G.F.R. study ("Birkoffer Report" [50]) have been performed in Canada. However, nearly thirty years experience in CANDU safety analysis and evaluation gives convincing proof that practically all relevant accident sequences have been analyzed to a degree which allows meaningful comparisons between canadian and foreign results.

The subsequent sections briefly describe the safety systems and their functions in a CANDU nuclear power plant, the present AECB safety goals, the actual performance of Ontario Hydro nuclear stations and the relative importance of human errors. The final section discusses the credibility of results derived from risk analysis.

6.3.2 Safety Systems *

The CANDU Nuclear Power Plant contains the following special safety systems :

- Shutdown System No. 1
- Shutdown System No.2
- Emergency Core Cooling System
- Containment System

*Material for this section has been taken from Background Paper No. 1 [30]. The detailed description refers to a CANDU 600 PHW.

as well as the following safety support systems :

- ° Emergency Water System
- ° Emergency Power System.

The special safety systems are independent in design and operation and free from operational connection with any of the process systems, including the regulating systems, to the greatest possible extent. Each special safety system is designed for an unavailability of less than 10^{-3} .

In order to protect the plant against "common mode" failures such as fires, turbine missiles, earthquakes and aircraft strikes that could affect safety and related systems at the same time, all systems have been divided into two groups. These groups are physically separated, so that common mode incidents either inside or outside the reactor building would not disable more than one of these groups.

Collectively, the basic functions of the two groups of systems are to :

- ° shutdown the reactor,
- ° remove decay heat from the reactor fuel,
- ° prevent any subsequent process failures,
- ° provide necessary information to permit operators to assess the state of the nuclear steam supply system.

a) Group one systems

Group one systems consist of the normal plant operating systems and the first line of defence systems. They are :

- Reactor Regulating System
- Shutdown System No. 1 (SDS1)
- Emergency Core Cooling
- All process systems except moderator cooling.

Reactor shutdown is carried out for Group 1 by the reactor regulating system or by shutdown system No. 1. Decay power is removed by discharge of steam from the boilers with make-up supplied by the auxiliary feedwater system.

b) Group two systems

Group two systems comprise systems that, acting together, will be capable of independent reactor shutdown, activity release prevention, and decay power removal. These systems are :

- Shutdown system number two (SDS2)
- Containment building (including dousing system)
- Auxiliary moderator cooling
- Emergency power and water supplies.

Reactor shutdown is carried out by shutdown system No. 2. Activity release is prevented by the containment system and also by attempting to maintain functional barriers after an accident. Decay power removal is affected by a number of means depending on the process failure and/or Group 1 system failure. A supply of emergency water is provided to the boilers for the purpose of removing decay power from any unfailed primary heat transport (PHT) loops. Decay power is removed from failed PHT loops by cooling the moderator.

Group one systems are supported by some 95 per cent of the total instrumentation and power requirements. Group two accounts for the remaining 5 per cent. The group one control area is the main control room, located in the Service Building adjacent to the Reactor Building. Cables exit the reactor building on one side and connect to the control equipment room. The second control area, which houses the Group two controls and instrumentation, is in the basement of the reactor auxiliary bay. Cables for this group leave the reactor building on a different side than the group one cables.

c) Shutdown system No. 1

Shutdown System No. 1 consists of 28 spring assisted gravity-drop shutoff rods and employs an independent triplicated logic system, which senses the requirement for reactor trip and de-energizes the direct current clutches to release the shutoff rods (Figure 12).

d) Shutdown system No. 2

Activation of the SDS2 results in the rapid injection of concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles. This second shutdown system employs an independent triplicated logic system which senses the requirement for activation and opens fast-acting helium pressure valves to inject the gadolinium poison into the moderator (Figure 13).

e) Emergency core cooling system (ECCS)

This safety system provides light water cooling to the reactor in the event of a loss of coolant accident which could result for instance in a large (say 20 cm diameter) pipe in the PHT system suddenly ruptured. The system operates in three stages (Figure 14).

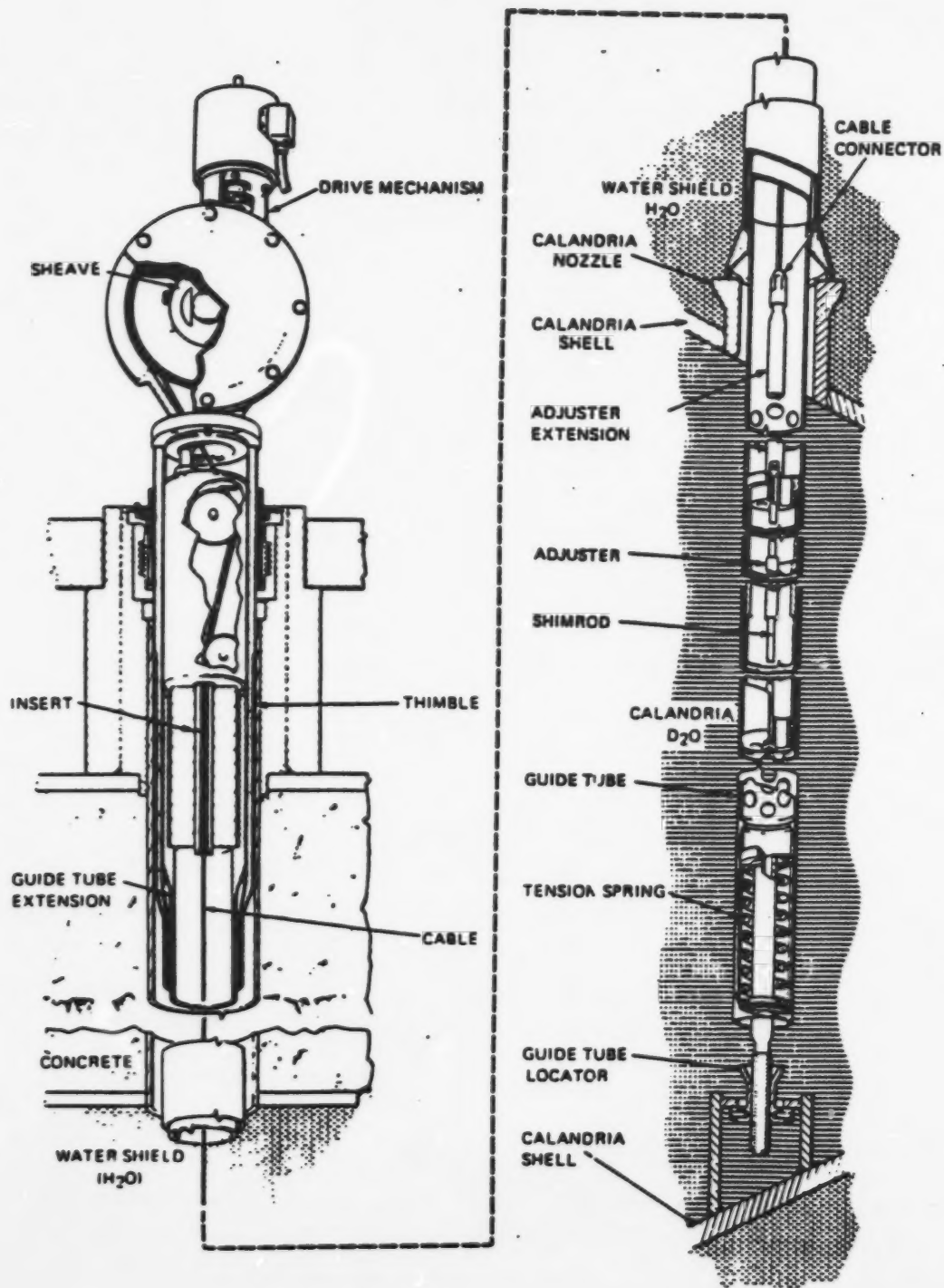


FIGURE 12

SHUTOFF ROD

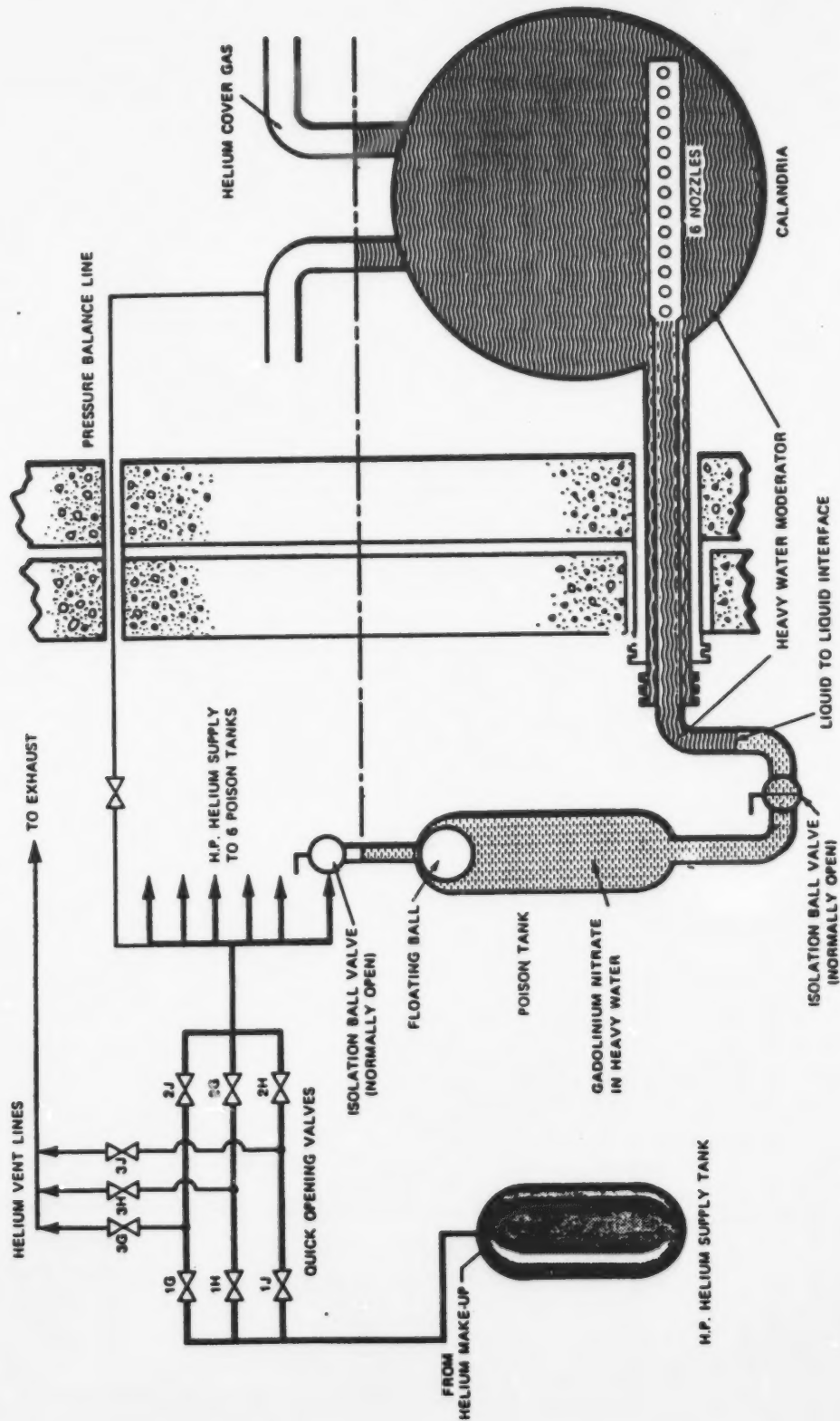


FIGURE 13 LIQUID INJECTION SHUTDOWN SYSTEM

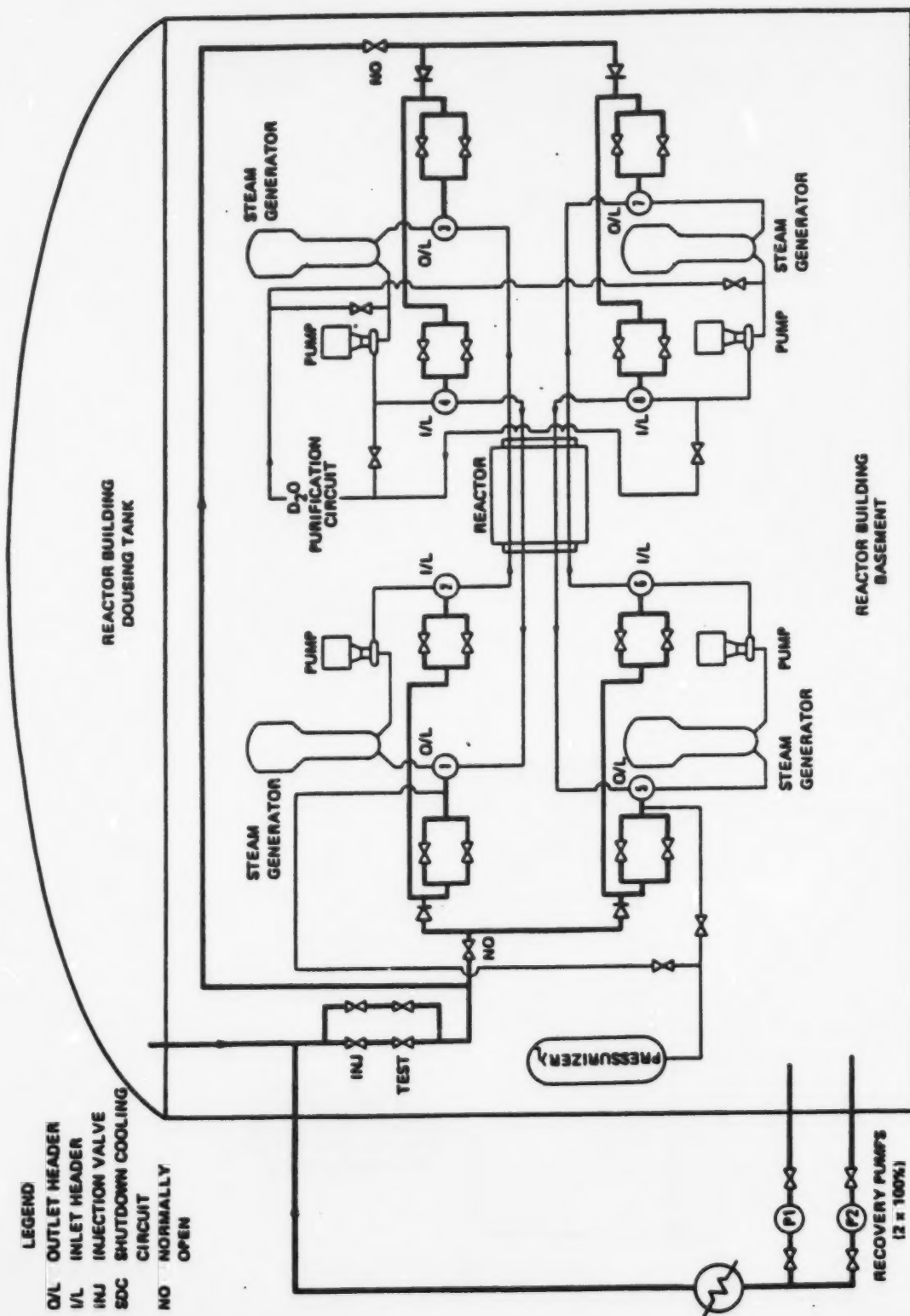


FIGURE 14 EMERGENCY CORE COOLING AND RECOVERY SYSTEM

The initial high pressure stage uses gas pressure to inject water into the reactor core from two water tanks located outside the reactor building. Following a loss of coolant accident the ECCS receives signals so that all injection control valves to the eight heat transport headers will open. The second stage starts when the water in the high pressure injection tanks is exhausted. At this time, valves at the outlet of the recovery pumps, which are an integral part of the ECCS, and valves in the line from the dousing tank open to supply water from the dousing tank via the recovery pumps. (The water stored in the bottom of the dousing tank is reserved for emergency core cooling and is not available to the dousing system for containment depressurization).

During the third stage, water injected in the first two stages is recovered from the basement of the reactor building of the reactor building and pumped back into the heat transport system. The recovery system includes two 100 per cent capacity pumps, mentioned above, and a heat exchanger for long term heat removal, which are located adjacent to the reactor building.

f) Containment system

The reactor containment building is the major component of this system. It is basically an envelope around the reactor and coolant systems where failure of these components could result in the release of significant amount of radioactivity to the public. Because of the large amount of energy stored in the reactor coolant systems, the envelope is capable of withstanding a significant pressure rise. The criterion for effectiveness of the envelope is the integrated leakage over the duration of the pressure excursion which would develop during a loss of coolant accident. To minimize the overpressure, an energy absorbing system comprises a source of dousing water, spray headers and initiating valves and building air coolers. Figure 15 is a schematic illustration of the containment system.

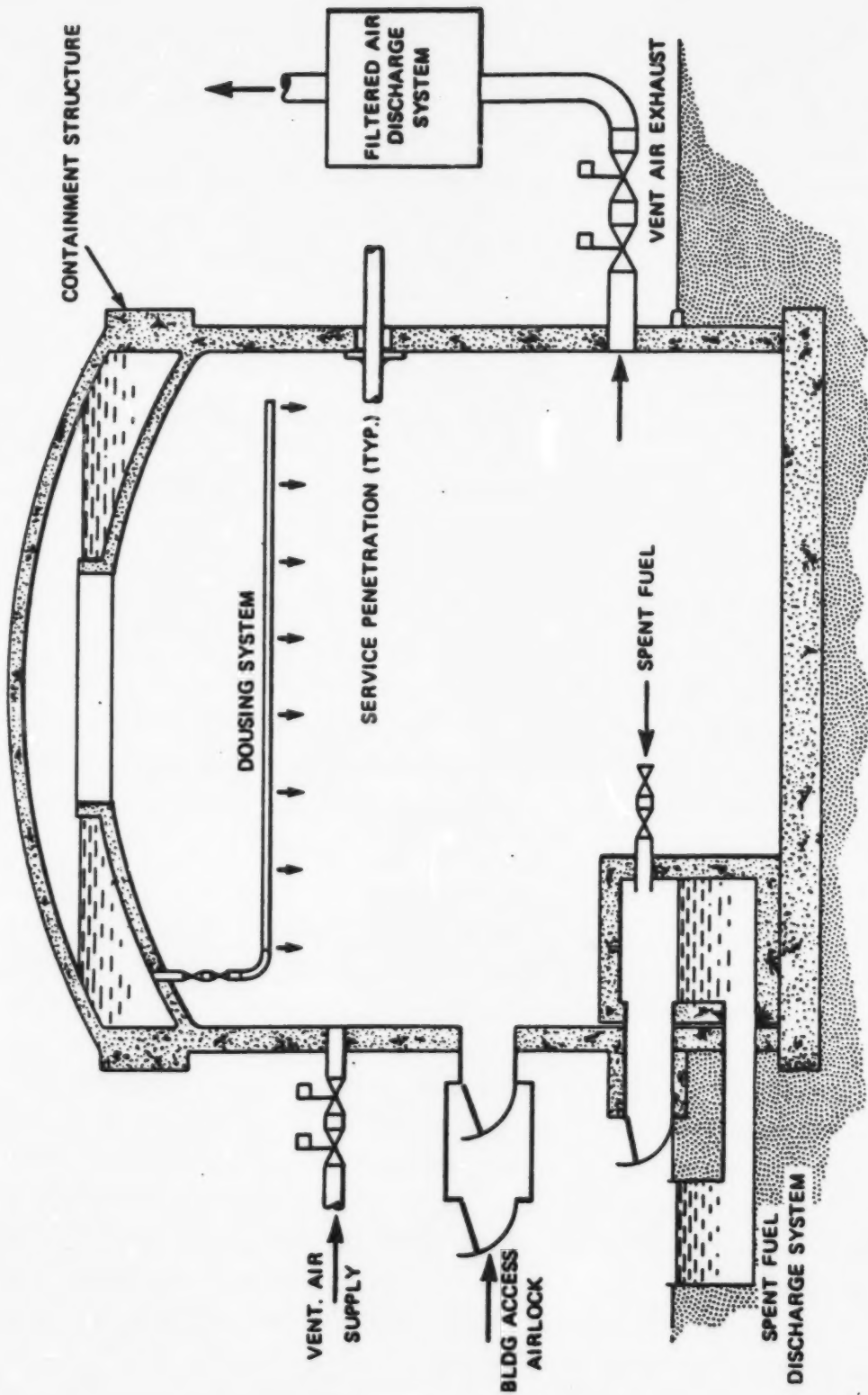


FIGURE 15 REACTOR CONTAINMENT SYSTEM

g) Emergency water supply

The system consists basically of two 100 per cent capacity diesel-driven main pumps which supply either demineralized water, or run cooling water from a separate intake structure. This water can be supplied to the boilers as feed water, to the primary heat transport system for make up, or to the moderator heat exchangers for cooling.

h) Emergency power supply

This system supplies the necessary power to the emergency water supply and to the Group two safety and control systems, for operator control of the station from the secondary control area. Switching loads from their normal supply to the emergency power supply is by means of manually operated breakers, so that the normal and emergency supplies cannot be connected at the same time.

6.3.3 AECB Safety Objectives*

A general safety objective endorsed by the AECB, which applies to all nuclear installations states that "the probability of malfunctions should be limited to small values, decreasing as the severity increases, so that the likelihood of catastrophic accidents is virtually zero" [51].

To ensure that such a low frequency could be achieved, it is required that process system failures ("single failures") are to have a total probability no greater than once in three years and are not to expose an individual in the general public at the site boundary to a total whole body dose exceeding 5×10^{-3} Sieverts, while process system failures combined with a failure of one of the special safety systems

*Material used in this section has been taken from [24].

("dual failures") are to have a total probability not greater than 3.3×10^{-4} per year, and are not to expose an individual in the general public to a whole dose exceeding 0.25 Sieverts. Failure of a process system and the simultaneous failures of two special systems ("triple failures") are ignored on the basis of the very low probability of such event sequences (3.3×10^{-7} per reactor year), considering the independence of the special safety systems.

Table 20, taken from [47] provides additional information, in particular with respect to maximum collective doses for single and dual failures. By using the above figures, one can infer that the maximum acceptable risks, for the most exposed individual, are 1.7 mSv/year and 0.1 mSv/year for single and dual failures respectively. The last figure corresponds to about one chance in a million per annum that the most exposed individual will suffer a premature death.

6.3.4 Safety Performance of CANDU Reactors*

According to reference [49] :

- During 80 reactor years of operation in Canada, there has never been a fatality nor has there been an injury of any kind for any reason to a member of the public.

- There has never been a release of radioactivity from any nuclear generating station which resulted in a measurable exposure to any member of the public.

*Material presented in this section has been taken from [49].

TABLE 20

OPERATING DOSE LIMITS AND REFERENCE DOSE LIMITS FOR ACCIDENT CONDITIONS

Situation	Assumed Maximum Frequency	Meteorology to be Used in Calculation	Maximum Individual Dose Limits	Maximum Total Population Dose Limits
Normal Operation		Weighted according to effect, i.e. frequency times dose for unit release	5 mSv/yr whole body 30 mSv/yr to thyroid	100 man-Sv/yr 100 thyroid-Sv/yr
Serious Process Equipment Failure (Single Failure)	1 per 3 years	Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete	5 mSv whole body 30 mSv to thyroid	100 man-Sv 100 thyroid-Sv
Process Equipment Failure plus Failure of any Special Safety System (Dual Failure)	1 per 3×10^3 years	Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete	250 mSv whole body 2500 mSv to thyroid	10^4 man-Sv 10^4 thyroid-Sv

- The predicted* radioactivity risk has met the risk corresponding to the AECB guidelines for every station for every year.

Lifetime results for Pickering are shown in Table 21. The

TABLE 21. Pickering NGS A Result Comparison (Lifetime)
System Versus Targets

	<u>Target</u>	<u>Results</u> <u>Frequency</u>	<u>Results</u> <u>Impairment</u> <u>Time</u>	<u>Results</u> <u>Inoperability</u>
<u>Process Systems</u>				
Regulating (LOR)	0.01/a	0.2/a		
Heat Transport (LOC)	0.001/a	0/a		
<u>Safety Systems</u>				
Shutdown	0.3%		0.05%	0%
ECI	0.3%		15.0%	0.2%
Containment	0.3%		6.0%	0%

difference between impairment and inoperability stems from the fact that, for reporting purposes, a special safety system is usually considered as unavailable when it is not 100 per cent effective, although it may still be able to fulfill its safety function adequately.

The predicted risk is of 0.002 mSv/year for single failures and of 0.26 mSv/year for dual failures. The last figure slightly exceeds the present AECB "risk criterion" but not the existing one when Pickering was licensed.

The safety system performance, in 1980, for all Ontario Hydro nuclear generating units is shown in Figure 16. This experience is typical of experience of other years.

*The predicted risk (to the most exposed number of the public) is calculated using all available equipment performance data.

FIGURE 16 .1980 SAFETY SYSTEM PERFORMANCE (INOPERABILITY)

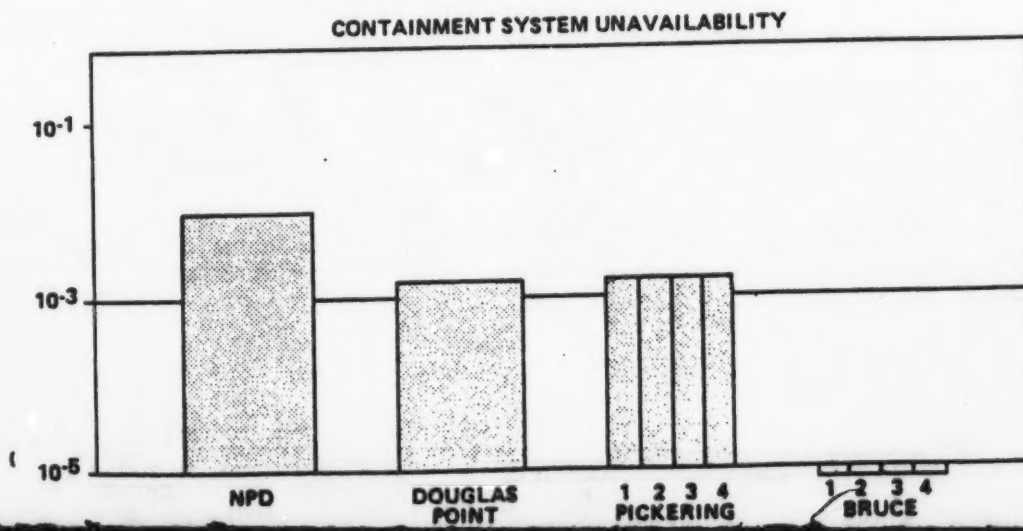
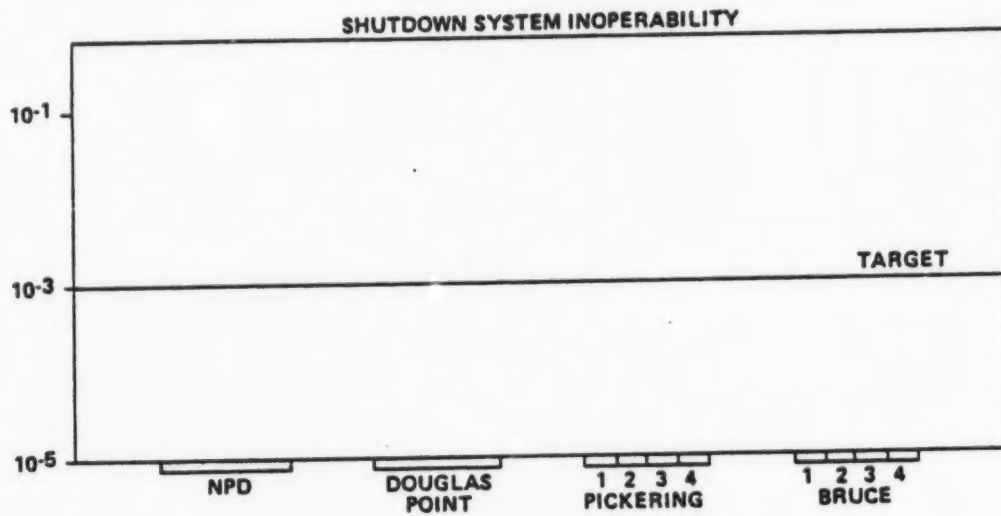
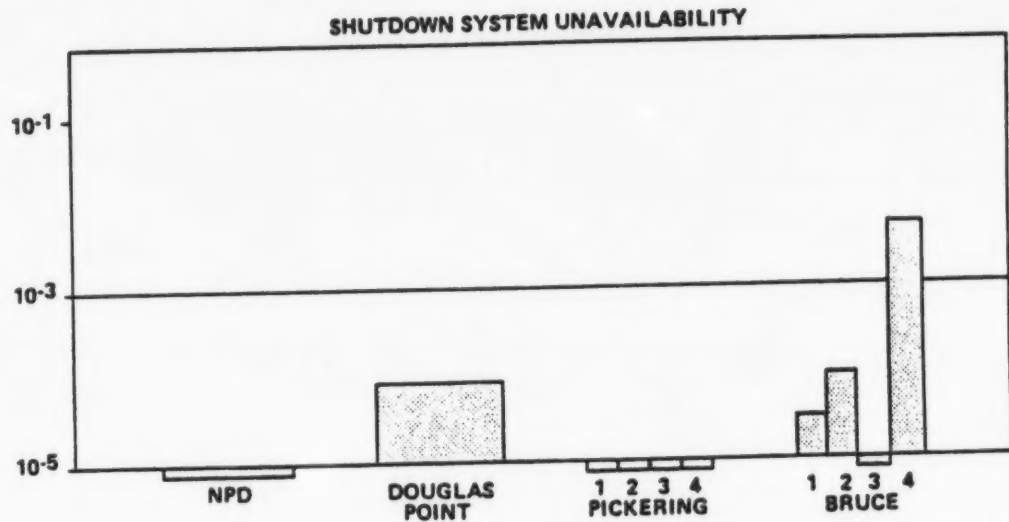
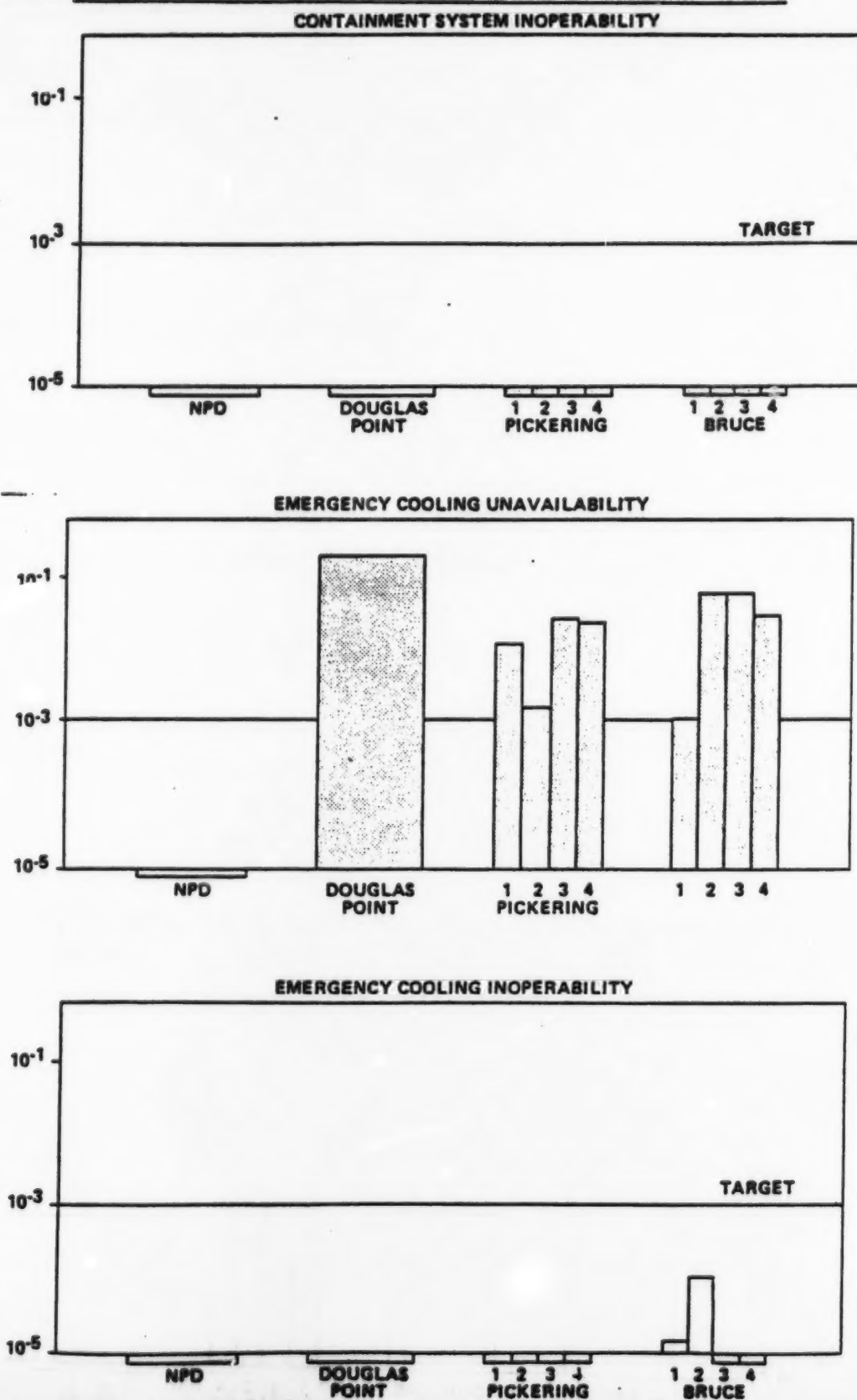


FIGURE 16 .1980 SAFETY SYSTEM PERFORMANCE (UNAVAILABILITY)



6.3.5 Relative Importance of Human Errors

A deliberate effort has been made by Ontario Hydro to investigate the importance of human errors in initiating or aggravating process failures and safety system unavailabilities. The results of this study have been reviewed by an independent task force [52]. Before interpreting the results, one should recall that due to a very limited number of recorded malfunctions, the statistics are poor. Here are the main findings :

- for serious process failures, the human error involvement is around 50% (within the 30% to 70% band).
- for safety system unavailabilities, the human error involvement is around 50% (38% for Pickering and 77% for Bruce), while the percentage of total impairment duration that can be attributed to human error is 33% (30% for Pickering and 95% for Bruce).

6.3.6 Credibility of Results in Risk Studies

Risk analysis of complex systems is a relatively new engineering branch which has been developed in aerospace and nuclear industry to prevent accidents of low frequency but of highly unacceptable results. Techniques such as fault tree analysis have gained international recognition and are now commonly used. They constitute an outstanding tool to systematically investigate potential system failures. Coupled with reliability theory, they provide the necessary scheme to estimate frequency of events, provided component failures are known. On the other hand, the understanding of physical processes which could occur during an accident in a nuclear power plant has also progressed considerably. As examples, one could cite the LOCA (Loss

of Coolant Accident) consequences in a CANDU reactor (i.e. the fuel behaviour and heat transfer to the moderator under various LOCA assumptions [53]) and the subsequent behaviour of the fission products within the containment (v.g. the partition coefficient of iodine under highly oxidizing conditions, such as air-saturated water).

These advancements have increased confidence in results obtained by risk analysis. Although it is generally agreed that risk estimates or particular sequences of events can be reasonably well predicted, doubts subsist concerning the ability of probabilistic methods in assessing total risk, since common mode failures, negligence and malevolent actions, not quantifiable in risk analysis, could also contribute to the overall risk. For these situations, however, special design requirements, such as the two-group approach indicated above (Section 6.3.2), strict procedural controls, etc., all contribute in minimizing the probabilities or the consequences of such occurrences.

There is therefore sufficient confidence that the target risk of about 0.1 mSv/year for the most exposed person, can be reached for low probability-high consequence postulated events. Best estimates for total risk lie within the 0.1-1.0 mSv/year range, i.e. a fatality risk of 10^{-5} - 10^{-6} per year.

At our knowledge, there are no published predictions in Canada on collective doses resulting from hypothetical accidents. Results obtained in U.S.A. are shown on Figure 17 [19] in which the complementary cumulative distribution risk function (see Section 2.4.c) is represented. It is generally considered, among the Canadian nuclear safety specialists, that the correspondent results for CANDUs would be better.

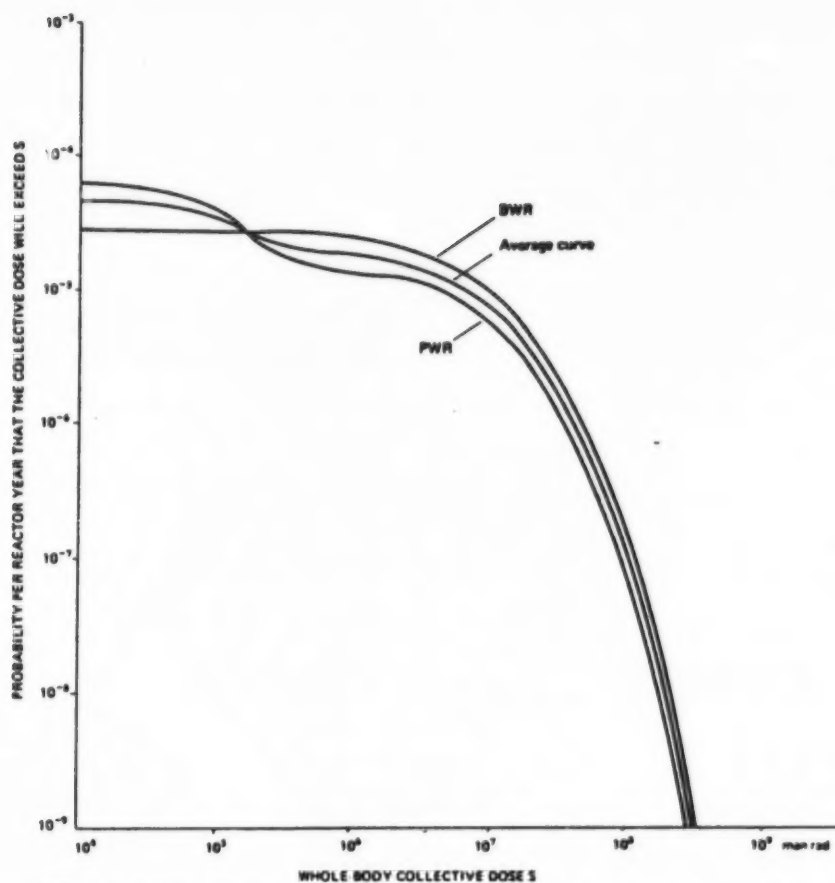


Figure 17. Probability distribution for the whole-body collective dose per reactor year. Reactor of 1000 MW(e). (Uncertainties can be accommodated by allowing variations in the average curve of factors of 1/3 and 6 in probability and 1/3 and 3 in consequence.) (155)

CHAPTER 7. WASTE MANAGEMENT*

7.1 Introduction

This chapter deals with radiological hazards resulting from managing reactor wastes and irradiated fuel. All reactor wastes made in Ontario are transported to the BNPD Radioactive Waste Operations Site for processing and storage. The spent fuel is stored at nuclear generating stations until such time that a geological disposal facility is available and/or a decision is taken on the reprocessing of irradiated fuel. Radiological risks have been calculated for the activities which take currently place and are under continuous assessment for irradiated fuel long-term storage, transportation and disposal. The following sections describe the control measures and the risk assessment for all these activities.

7.2 Reactor Wastes

7.2.1 Storage

All facilities are designed to confine their stored radioactivity over a design lifetime of 50 years. In-ground structures are serviced by surface and subsurface drainage systems to allow prompt detection of any escaping radioactivity. Groundwater sampling wells, located at various distances from the facilities, provide additional surveillance. All storage structures are serviced by a program of regular inspection and maintenance.

*Material for this chapter has been mainly taken from [41]. A detailed discussion on spent fuel disposal and pathway analysis can be found in Background Paper No. 4 [31].

Chronic emissions (mainly tritium) are generally very small, not exceeding 10^{-2} mSv/a for the most exposed individual from the public. Exposures from accidental releases should be less than 5 mSv; in the case of fire in the warehouse structure, the maximum dose was estimated to be less than 0.7 mSv/event.

7.2.2 Processing

The exhaust from all reactor waste processing equipment is filtered, reducing particulate emissions to insignificant levels. Charcoal filters are not employed since radioiodine levels are very low as a result of handling delays prior to processing. Gross dose-rate guidelines for permitting processing provide additional protection in this respect. Exhausts are continuously sampled for tritium, particulate and radioiodine, and environmental monitoring surveillance is provided by the array of BNPD Environmental Monitoring Sites.

Chronic emissions (mainly tritium) are estimated to result in maximum individual dose of 5×10^{-3} mSv/a. Accidental releases (from a hypothetical major fire) would not result in doses larger than 0.2 mSv/event.

7.2.3 Transportation

Both low level reactor wastes and tritiated heavy water are transported in IAEA Type A containers which are designed to remain intact during normal handling but not during accidents. Potential accident consequences are limited by regulatory inventory restrictions. The medium level reactor wastes, such as filters and ion exchange

columns, are transported in IAEA Type B containers, which are designed to prevent the release of the contents during reference accident conditions. There is no release of radioactivity during normal shipment. Accidents, involving tritiated heavy water shipments, would be expected to result in the largest potential public dose. Estimated maximum individual risk ranges from 6×10^{-5} to 4×10^{-3} mSv/a, according to different meteorological conditions and spill severities.

Until now, there has been no accidental releases of radioactivity during storage, processing or transportation activities in Canada. Table 22 summarizes the estimated reactor risks.

TABLE 22. Reactor Waste Risks

	<u>Chronic</u> mSv	<u>Acute</u> mSv/event
Storage	1×10^{-2}	5
Processing	5×10^{-3}	2×10^{-2}
Transport	-	6×10^{-5} - 4×10^{-3} **
Disposal	-	-
Decommissioning	1×10^{-1} *	-

* Chronic public risks during decommissioning are not expected to exceed those of a normal operating station

** Based on D₂O tritium concentration of 1 Ci/kg

7.3 Irradiated Fuel

The irradiated fuel contains more than 99.99% of the total radioactivity produced in a nuclear generating station. Specific activities (in curies/kg U) for the most relevant fission products and actinides in an irradiated fuel bundle are shown in Table 23, at discharge, after one year and after 10 years.

TABLE 23

ACTIVITIES OF SELECTED FISSION PRODUCTS AND ACTINIDES IN IRRADIATED FUEL*

Radionuclide	Mode of decay	Half-life (Years)	Activity (curies/kg U)		
			at discharge	after 1 year	after 10 years
Fission Products					
Tritium (H-3)	β	12.3	0.17	0.16	0.10
Krypton-85	β,γ	10.7	2.22	2.19	1.23
Strontium-89	β,γ	0.14	443	3.95	9.8x10 ⁻²⁰
Strontium-90	β	29	17.5	16.0	12.9
Yttrium-91	β,γ	0.16	578	77	1.1x10 ⁻¹⁶
Zirconium-95	β,γ	0.18	825	17.3	1.3x10 ⁻¹⁴
Niobium-95	β,γ	0.10	802	36.6	2.9x10 ⁻¹⁴
Technetium-99	β,γ	2.1x10 ⁵	3.4x10 ⁻³	3.4x10 ⁻³	3.4x10 ⁻³
Ruthenium-106	β	1.0	182	101	0.21
Iodine-129	β,γ	1.6x10 ⁷	7x10 ⁻⁶	7.9x10 ⁻⁶	7.9x10 ⁻⁶
Iodine-131	β,γ	0.02	525	1.2x10 ⁻¹¹	0
Cesium-134	β,γ	2.17	16.9	11.3	0.55
Cesium-135	β	2.3x10 ⁶	4.5x10 ⁻⁵	3.8x10 ⁻⁵	3.8x10 ⁻⁵
Cesium-137	β,γ	30.2	25.3	24.8	20.2
Cerium-144	β,γ	0.78	424	181	0.06
Promethium-147	β,γ	2.6	58.9	50.7	4.7
Actinides					
Neptunium-237	α,γ	2.1x10 ⁶	2.1x10 ⁻⁵	2.1x10 ⁻⁵	2.2x10 ⁻⁵
Plutonium-238	α,γ	87.7	7.2x10 ⁻²	8.3x10 ⁻²	8.0x10 ⁻²
Plutonium-239	α,γ	2.4x10 ⁴	0.15	0.15	0.15
Plutonium-240	α,γ	0.24	0.24	0.24	0.24
Plutonium-241	β,γ	14.7	22.9	21.8	14.2
Americium-241	α,γ	432	11.5x10 ⁻³	4.7x10 ⁻²	0.3
Americium-243	α,γ	7380	5.3x10 ⁻⁴	5.3x10 ⁻⁴	5.3x10 ⁻⁴
Curium-242	α,γ	0.45	2.58	0.44	8.9x10 ⁻⁶
Curium-244	α,γ	18.1	1.6x10 ⁻²	1.5x10 ⁻²	1.1x10 ⁻²

* Based on Pickering fuel irradiated to 7.5 MWd/kg U⁽²³⁾.

7.3.1 Storage

Most of the radioactivity in the irradiated fuel is confined within the uranium oxide matrix and enveloped by the zircaloy sheath of the fuel elements. Although the total inventory of long lived radio-nuclides in the irradiated fuel bay is large, releases of radioactivity into the bay water during normal operation is very small, since the occurrence of fuel sheath defects has been infrequent and defective fuel is canned before storage, if necessary. A purification system maintains radioactivity in the bay water at a very low level, while airborne radioactivity is controlled by a ventilation system employing high efficiency particulate and charcoal filters.

Collective doses resulting from chronic noble gas releases have been estimated out to a distance of 100 km from the storage site and for different siting scenarios. The maximum collective whole body dose was found to be less than 10^{-2} person-mSv/a.

A variety of accidental releases have also been analyzed. The most restrictive accident identified was the dropping of a fuel module, i.e. a container used for storing fuel bundles, containing freshly irradiated fuel during fresh fuel transfer to an auxiliary bay. The maximum estimated risks are 3.5×10^{-4} mSv/a (thyroid) and 2.1×10^{-4} mSv/a (whole body).

7.3.2 Transport

To obtain a fuel flask shipping licence in Canada, an operator must show that the flask meets or exceeds IAEA qualification test requirements. These tests, which include drop tests and a fire test at 800°C for 30 minutes, must demonstrate that the flask can maintain its shielding and leak tightness integrity.

The individual dose (via direct radiation) to a member of the public residing close to a shipment route has been conservatively assessed to be less than 8×10^{-6} mSv/a. The estimated collective dose would not exceed 300 person-mSv/a.

There are no independent Canadian estimates on risks from accidental releases. By using information from two US studies [53, 54], it has been estimated that the probability of an accident of worst consequences (30 mSv) is less than 2×10^{-7} /y; the individual radiological risk would then be less than 6×10^{-6} mSv/a.

7.3.3 Disposal

At the end of 1980, about 4400 tons of spent fuel was already stored in water-filled pools at the end of 1980 [55] and nearly 1000 tons are annually added. In order to safely confine this accumulating nuclear fuel wastes, the federal government has given to AECL the responsibility of developing a disposal program with the overall objective to ensure that there will be no significant effect on man or the environment at any time.

The program has been divided up into three phases - concept assessment, site selection and construction and operation of a demonstration disposal facility.

a) Concept assessment

The concept consists of isolating the radioactive materials by disposal of immobilized wastes in a stable rock formation*. Plutons are the geologic formations currently being investigated. Most of the

*This appears to be the most attractive approach; other options, such as disposal in bedded salt, clay or shale formations, are left open.

concept assessment phase is thus directed toward the research on geological, hydrological and chemical behaviour of plutonic rocks. Once the concept has been accepted as sound (by various independent committees of experts), the program proceeds to the next phase, the site selection.

b) Site selection

After selection of a number of technically suitable sites has been made, based on the results previously obtained, this phase will explore the social and political considerations associated with these suitable sites.

c) Construction and operation of a demonstration facility

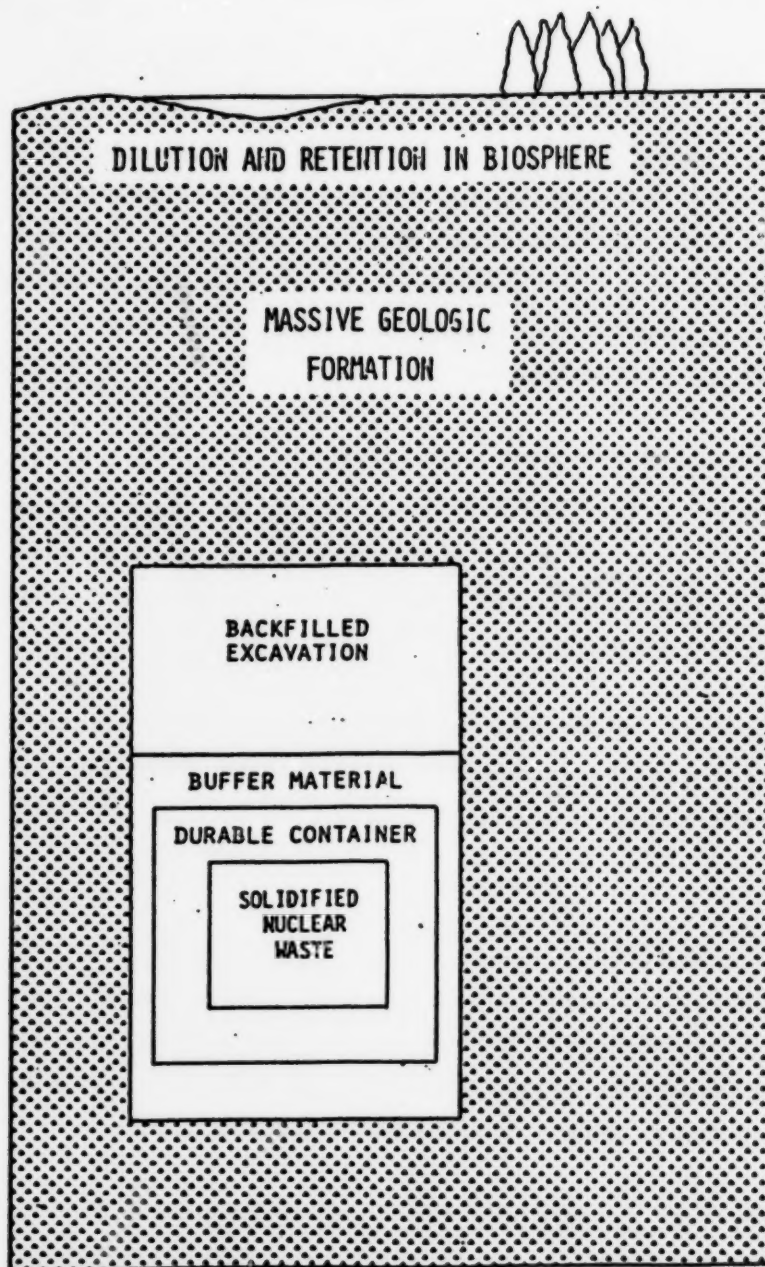
The full-scale facility cannot be built before the site chosen has met completely the expected design characteristics. A demonstration facility will be constructed in order to measure the physical characteristics of the rock and to test the equipment and handling methods. Immobilized wastes would be emplaced with an extensive monitoring program if the previous tests confirm the design of the facility could be built by expanding the demonstration facility.

7.3.3.1 Safety Principles

The safety principles guiding waste fuel management are similar to those who apply for preventing adverse consequences from reactor accidents : the potentially harmful radionuclides are to be kept away from human environment by means of a number of barriers (Figure 18).

First, the UO_2 matrix forms a barrier because it holds most of the radionuclides. Then the zircaloy sheath, because it protects the

FIGURE 18



FEATURES PROTECTING MAN FROM NUCLEAR WASTE

From Ref. 56.

matrix, is the second barrier. By filling the surrounding spaces around the container with bentonite, the radionuclides have to force a fourth barrier. So far, these barriers are man-made; they are also called engineered barriers. The natural barriers are the geosphere and the biosphere. By choosing an appropriate site, the properties of the natural barriers can be optimized.

7.3.3.2 Research and development activities

Conceptual designs for a disposal vault for the emplacement of either immobilized fuel or immobilized reprocessed waste are now complete [57, 58]. These facilities could be constructed on the basis of presently available technology.

Most of the research and development program is devoted to predicting how radioactive material might escape from the disposal vault and migrate through the geosphere and biosphere to man over long-time periods.

a) Geotechnical research

Geophysical, geological and hydrogeological studies have been done extensively at Forsberg Lake and in CRNL and WNRE areas. Work has started at Lac du Bonnet region, northeast of WNRE site, in view of a possible siting for an underground research laboratory.

The physical properties, the integrity and the structure of rock bodies are characterized by a variety of geophysical methods such as airborne magnetic gradiometer and electromagnetic surveys, ground measurements of density and seismic reflection studies, etc. Rock characteristics, fractures, zones of alteration and rock quality can be found by interpreting measurements obtained in bore-holes.

From surface and subsurface data, lithological and fracture models for pathways analysis and for rock mass comparisons can be made. Therefore, studies such as three-dimensional modelling of open fractures, as determined from boreholes measurement are under way.

Characteristics of groundwater such as flow direction and groundwater chemistry are extremely important to analyze. In fact, the assessment of the safety of geologic disposal is closely dependent of predicting water flow. Work on flow modelling, by using both equivalent porous media and fractured media, is progressing.

Various measurements made on drill-core samples are determining the physical, mechanical and thermal properties of hard-rock formations. An underground research laboratory, presently under construction in the Lac du Bonnet batholith near WNRE, will permit the performance of experiments done on scale and in an environment similar to the planned full-scale disposal vault. The laboratory will consist of several rooms at a depth of about 300 metres. Information on geological, hydrogeological and geochemical characteristics of the rock mass will be obtained. The effects of excavation will also be assessed.

b) Applied chemistry and geochemistry

The interactions of groundwater with the waste form and the geologic formation are being studied in order to evaluate the behaviour in the vault environment. Developments of backfill and buffer materials which can retard the entry of water and slow down the flow of leached radionuclides are under progress.

c) Pathway analysis

Figure 19 [41] shows a generalized flow chart of possible exposure pathways to man. Existing computer codes are being used to assist the environmental impact of radioactive materials released before the final depository closure.

The post-closure assessment methodology is indicated in Figure 20 [42]. Detailed computer programs are being developed and applied for hydrogeological and chemical modelling, while a "systems variability analysis" code (SYVAC [59]) integrates the total system and samples data from distributions reflecting the uncertainty and variability in the data values. The resulting output is a histogram of consequences (dose to man) versus probability, indicating the most probable sequence of events, and other consequence estimates, together with their probability of occurrence.

Figure 21 illustrates the application of computer programs to model the flow of groundwater in the vault and in the surrounding geological formation, through both porous and fractured media. Figure 22 illustrates the approach to chemical modelling which includes container corrosion, waste dissolution and mass transport with chemical retardation through buffer, backfill and fractured rock. Three sub-models describe the vault, the geosphere and the biosphere; for the latter, two options are considered : arrival of radionuclides at a valley bottom or directly into a lake.

d) Risk assessments

The assessment methodology used in SYVAC is indicated in Figure 20. Model parameters are represented as distributions rather than single values, to allow for uncertainties in these values. Sampling a value of each parameter from its distribution defines a "scenario". The computer program then calculates the consequences of this scenario,

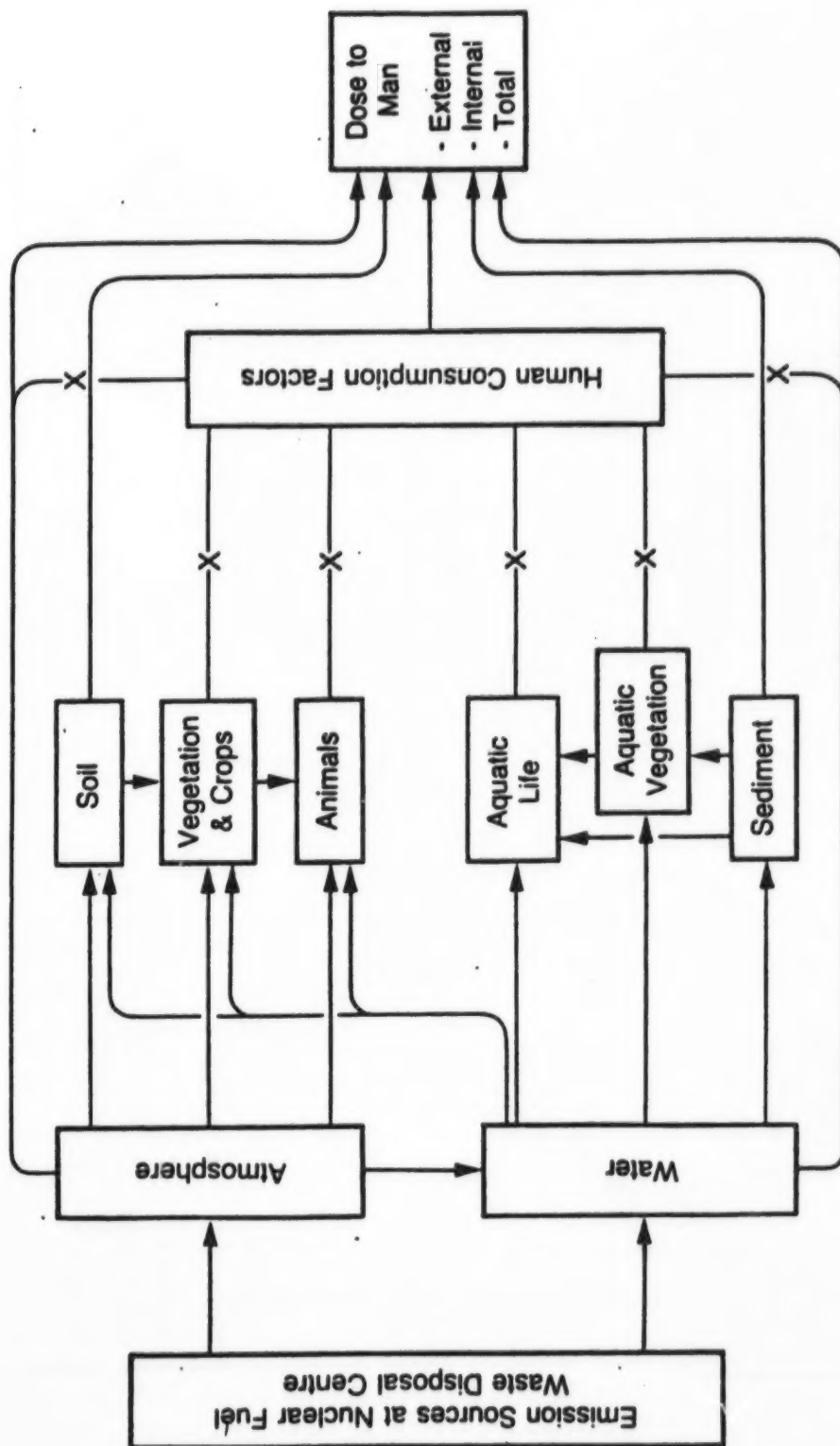


FIGURE 19
GENERALIZED FLOWCHART OF POSSIBLE
EXPOSURE PATHWAYS TO MAN

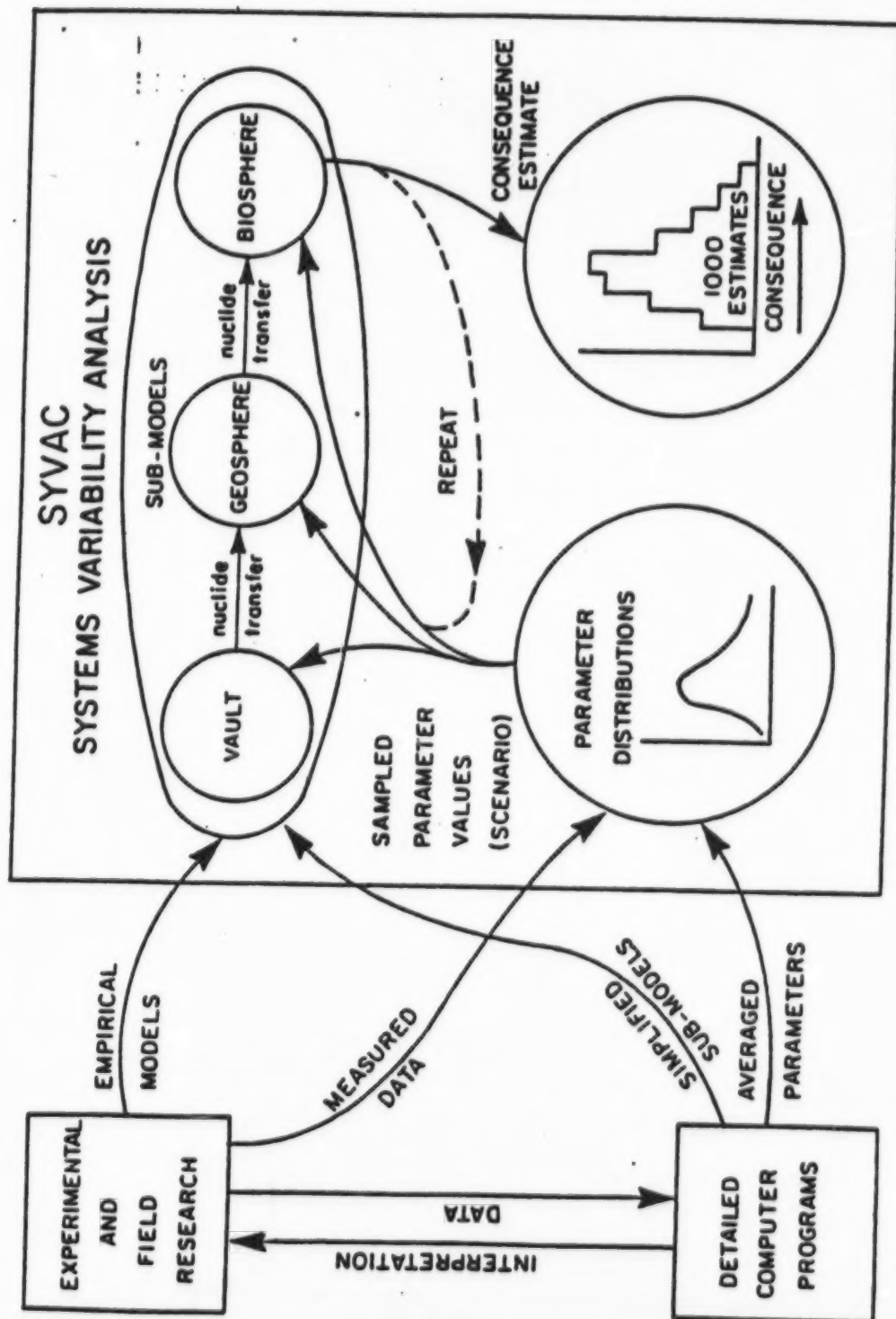


FIGURE 20
POST-CLOSURE ASSESSMENT METHODOLOGY

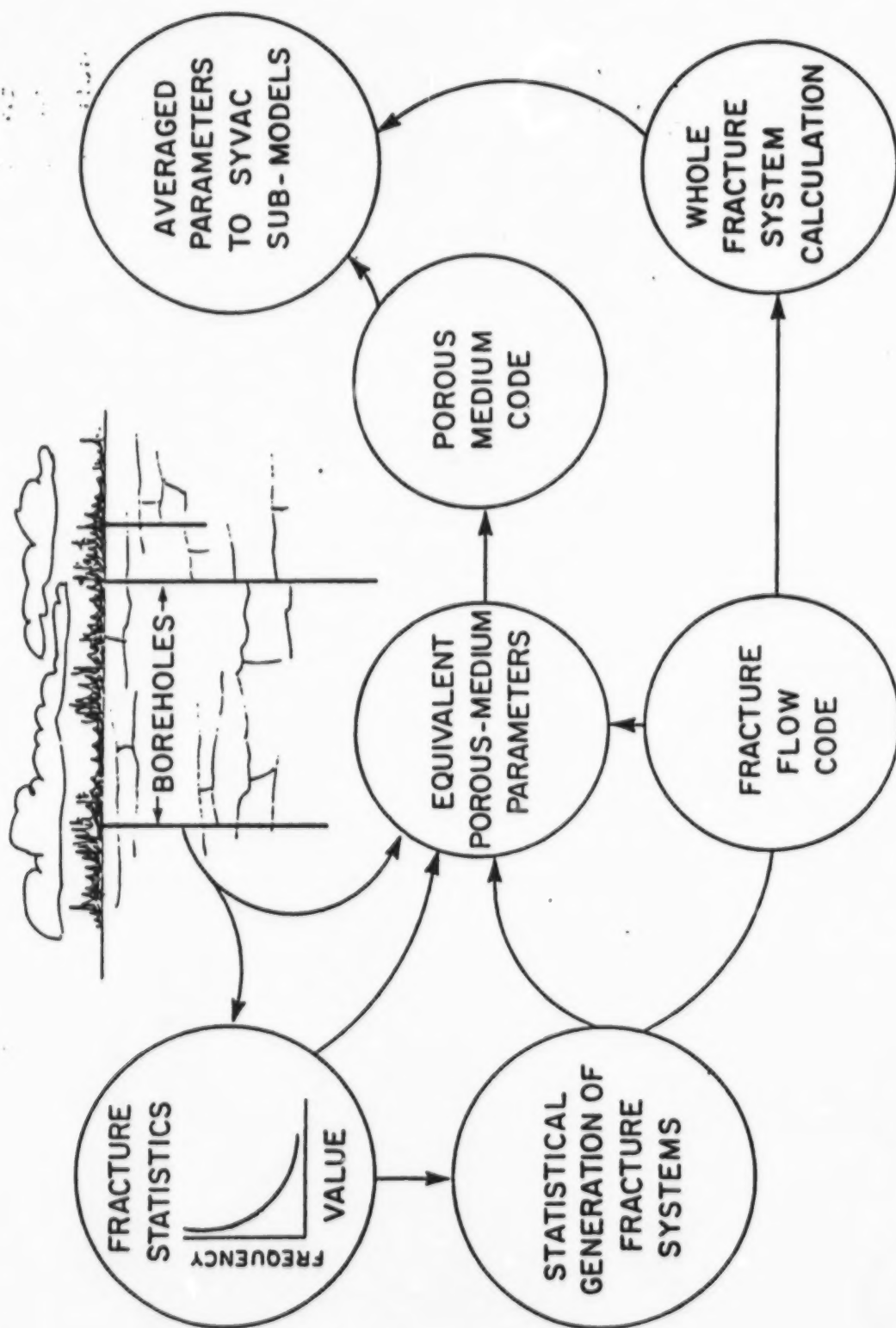


FIGURE 21
HYDROGEOLOGICAL MODELLING

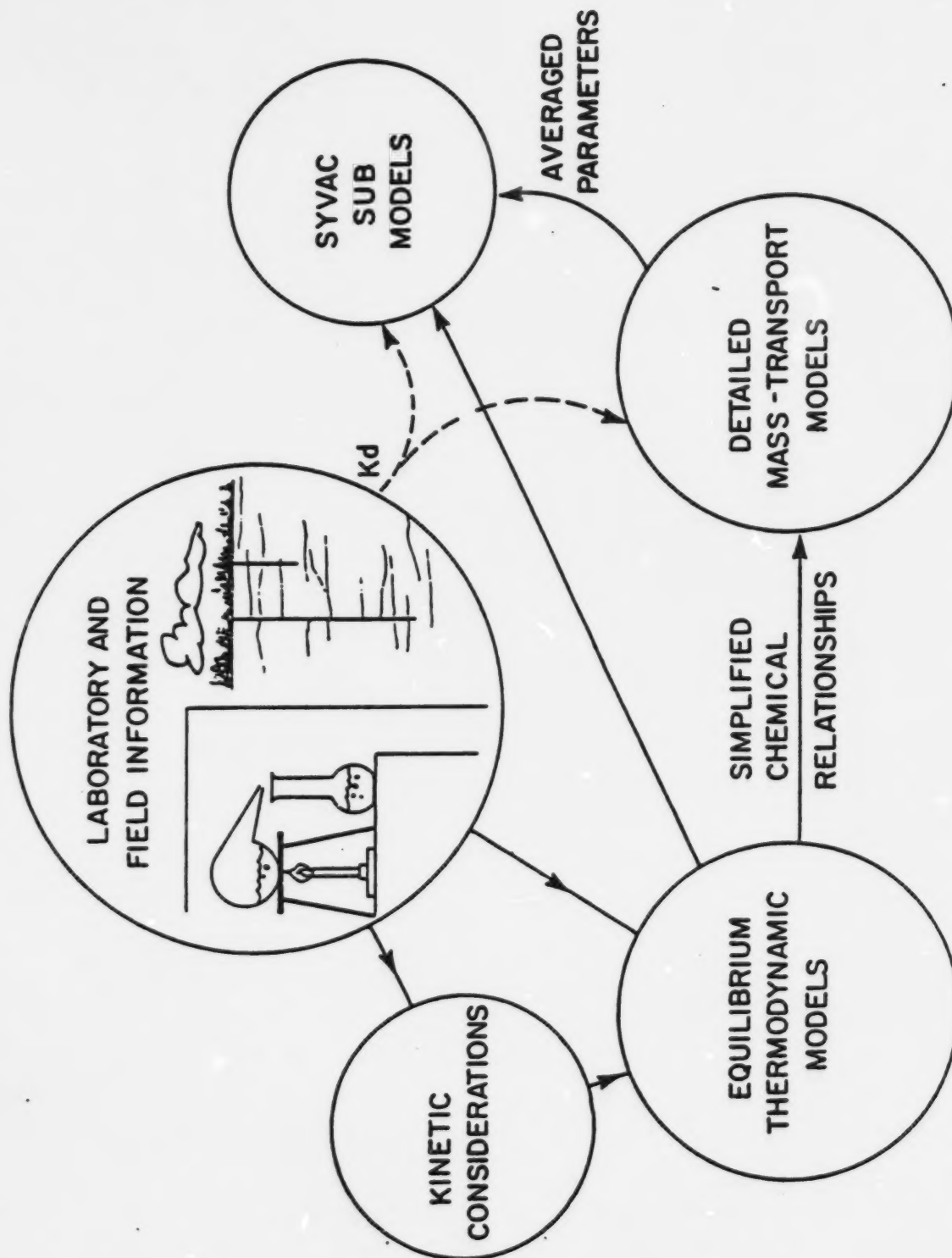


FIGURE 22
CHEMICAL MODELLING



usually the maximum dose to an individual. By repeating this procedure, one obtains a histogram of consequence estimates versus frequency of occurrence. As examples, Figure 23 shows results of a single run, for a particular scenario, whereas Figure 24 gives the results from over 3000 runs. Integration time was one million year.

e) Comments

At the present stage, these results should be considered only as preliminary, although work done in U.S., U.K. and Sweden point to the same low-risk outcome (Figure 25). The models are still relatively simple and effects of drastically changing geological conditions have still to be assessed. At this time, results presented in the last three figures still represent the best estimate of possible long-term radiological impacts from nuclear wastes.

7.3.4 Summary

The radiological risks estimates resulting from storage, transport and disposal are summarized in Table 24.

Table 24

IRRADIATED FUEL RISKS

	Chronic mSv/a	Acute
Storage	$< 2 \times 10^{-2} (C)$	$3.5 \times 10^{-4} (I. Th)^*$ $2.1 \times 10^{-4} (I. WB)^*$
Transport	$< 300 (C)$ $< 8 \times 10^{-4} (I)$	$< 6 \times 10^{-6} (I)$
Disposal		
Preclosure	No Data	
Postclosure	See Preliminary Probability Distribution (Figure 24)	

* - During bay relining period only

I - Individual Dose

Th - Thyroid Dose

C - Collective Dose

WB - Whole Body Dose

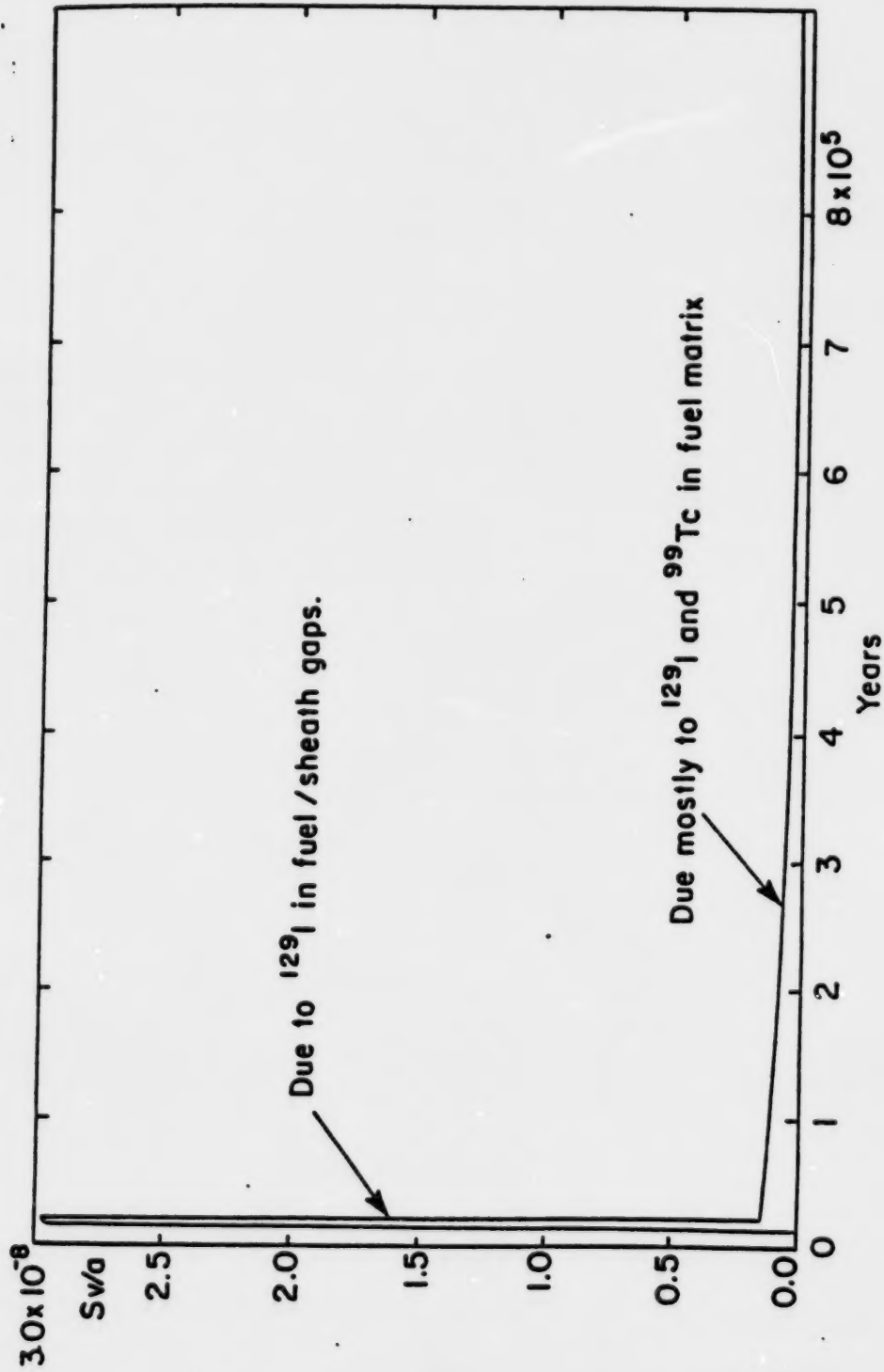


FIGURE 23
SINGLE CASE -- DOSE VS TIME

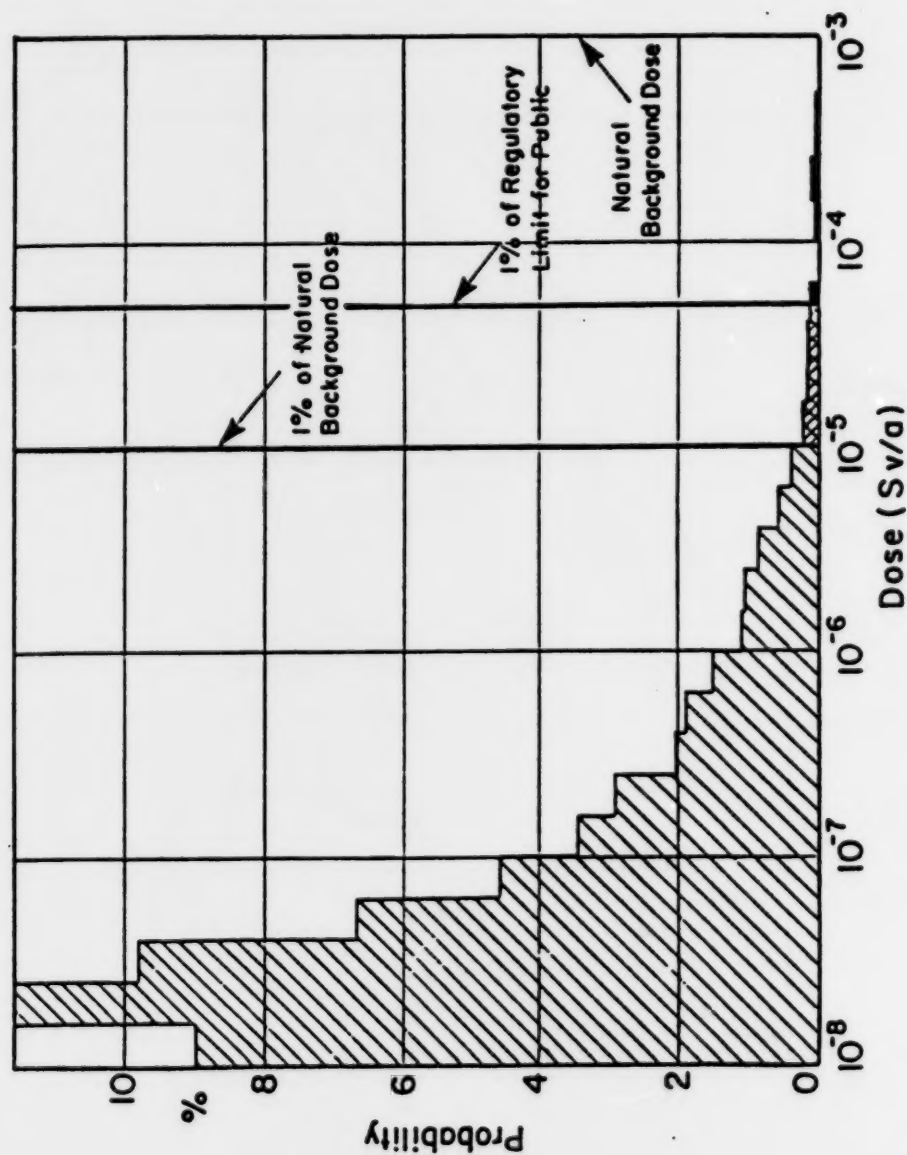


FIGURE 24
MAXIMUM DOSE TO MOST EXPOSED INDIVIDUAL

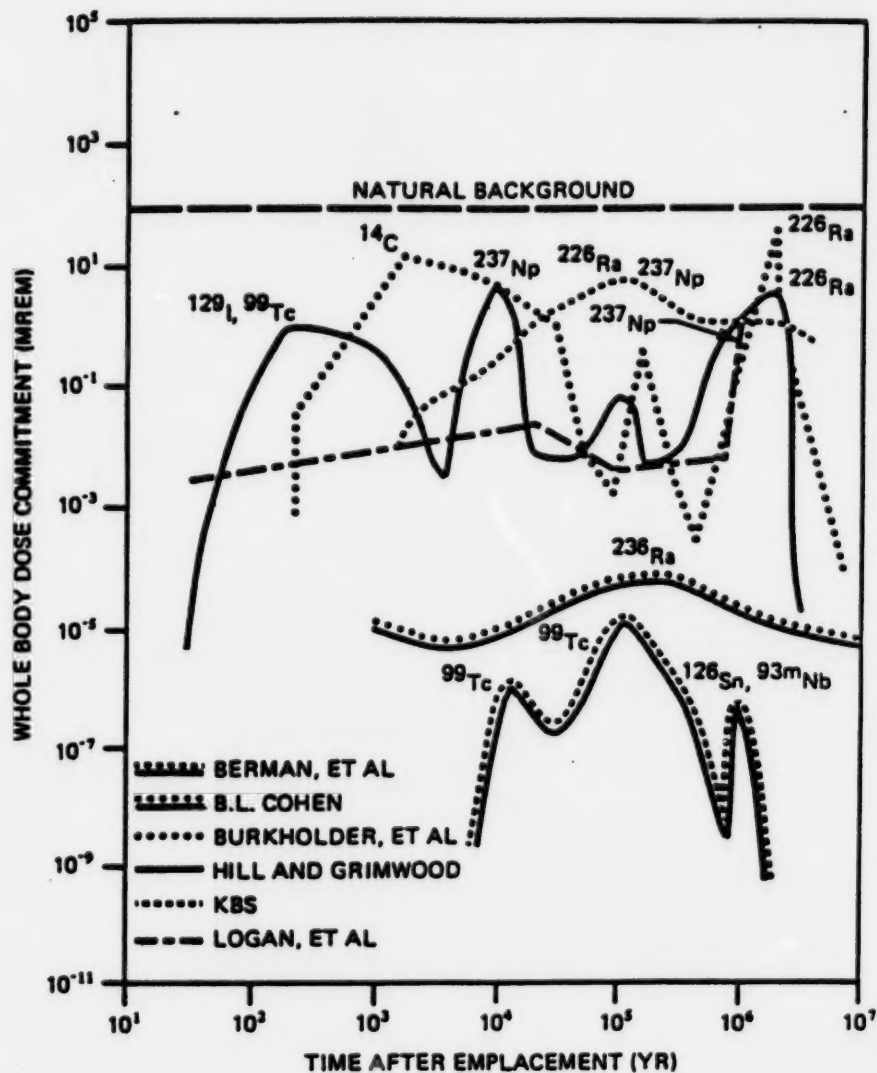


FIGURE 25
POTENTIAL INDIVIDUAL DOSES
(TASC SAFETY ASSESSMENT REVIEW)

CHAPTER 8. CONCLUSIONS

A deliberate attempt has been made in this report to collect, in a single document, information essential to radiological risk assessments. Thus, physical processes involving the handling of radioactive materials, emission point sources, control measures taken to keep chronic radioactive releases to a minimum, protective features designed to minimize the probability and the consequences of accidents, and assessment techniques employed to calculate maximum individual doses and probable collective doses, have been described and analysed for each activity within the CANDU nuclear fuel cycle.

This information has been extracted from more than three hundred references, indicated in the Background papers. As mentioned in the foreword, part of the material which has been reviewed has not been incorporated in this report, partly because of lack of time but also to keep the report at a manageable size. Also, some information is outdated. It is, however, believed that additional or new information would not significantly change the main findings in this report.

What are these findings ?

First, considerable work in assessing radiological hazards from chronic releases has already been done.

Second, for accidental releases, probabilistic methods of assessing individual and collective doses resulting from activities along the nuclear fuel cycle are becoming increasingly accurate and, as a consequence, they are being increasingly used to ensure that the safety objectives are being met. These methods are also being used to

evaluate alternative design or management techniques to decrease present and future radiological hazards resulting from uranium mill tailings, from reactor operation and from waste management.

Third, maximum individual doses, for members of the public, arising from normal radioactive releases, are kept to low levels (at less than 1% of the regulatory limits) for all activities within the nuclear fuel cycle.

Fourth, maximum individual risks (i.e. the probability of life reduction by cancer) arising from hypothetical accidents, are considered not to exceed 10^{-5} per year, per activity.

Fifth, collective doses are estimated to be very small, up to about 3 person-Sv. The committed doses could be hundred times higher but the corresponding risk remains negligible with respect to many other risks, in particular with risks resulting from background radiation.

Sixth, more work is needed to obtain in a tabulated form, normalized risks, i.e. risks relative to a unit energy production, for all activities within the nuclear fuel cycle.

Seventh, numerical estimates made for the CANDU fuel cycle are in agreement with or lower than the corresponding estimates made for the PWR fuel cycle.

Finally, the radiological risks from nuclear power are considerably lower than the non-radiological risks as assessed in Dr. Zikovsky's separate report.



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